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Prototype "Monju" Fast Breeder Reactor

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Mission and Development Process

94FE0753A Tokyo GENSHIRYOKU KOGYO in Japanese Jun 94 pp 12-16

[FBIS Translated Text]

1. "Monju" Missions

"Monju" is a plant which is at the stage just prior to the demonstration and practical reactors needed to put fast breeder reactors (FBRs) into service. As such it has the following missions.

(1) The experimental reactor "Joyo" has systems, equipment configuration, and plant scale designed to demonstrate the basic performance and safety of FBR plants in terms of reactor core and sodium cooling system. Monju, on the other hand, at the prototype reactor stage, is characterized by being a power-generating plant equipped with steam generators and other power-generating equipment, and by being roughly seven times larger than Joyo in terms of heat output.

Accordingly, Monju's missions involve the demonstration of the performance, safety, and reliability of an FBR which generates power, through the experience gained in its design, fabrication, construction, and operation. Also, with the experience gained in operating Monju, technological developments will be pursued to further improve safety and reliability, to enhance the operating rate which is connected to economy, and to better the operating performance. These benefits will then be reflected in the development of the demonstration and practical reactors which are to come after Monju.

(2) In developing Monju, R&D was conducted in a wide range of fields to overcome the engineering problems peculiar to FBRs. This R&D was conducted by the Japan Atomic Energy Research Institute (JAERI), primarily through the Power Reactor and Nuclear Fuel Development Corporation (Donen), but also with the cooperation of numerous Japanese universities, electric power companies, and manufacturers. The design, equipment, system fabrication, and plant construction all reflect these R&D efforts, and their findings and successes. Thus the development of basic, fundamental FBR technology in Japan is another major mission of Monju.

2. Monju Development and Program History

2.1 Monju Development History

(1) Overview

The Atomic Energy Commission drafted, in 1966, the report entitled "Re Basic Policies on Power Reactor Development." In this report, the development of the FBR was taken up as a national project, and the decision to move ahead with development was made. In the wake of this, Donen was established, and development work began. The objective was a sodium-cooled fast breeder reactor using a mixed fuel of plutonium-uranium oxides. An experimental reactor was to be followed with a prototype reactor, developed in Japan, having a power output of between 200,000 and 300,000 kW.

In the design of Monju, following this objective, the major parameters such as fuel design conditions, cooling system temperature, and steam conditions would be made close to the conditions found in a practical reactor. In selecting the fuel, reactor structure, fuel converters, steam generators, and other major equipment, the transferability of these to future practical reactors was taken into consideration. The basic design specifications of Monju were determined after studies and evaluations of the preliminary design which began in 1968.

(2) Design History

The basic conceptual design of Monju was solidified following the beginning of preliminary design work in 1968. Based on this, application was made for a nuclear reactor installation permit in 1980. In the interim, many parameter surveys were conducted, and the Monju plant design has been optimized stage by stage.

The specification items studied and evaluated at the basic design stage included the following major ones.

(1) Nuclear reactor type

(2) Fuel exchange method

(3) Dimensions of and number of fuel pins in fuel aggregates

(4) Sodium coolant temperature, steam conditions

(5) Reheating, non-reheating cycle

(6) Main circulating pump positions in primary and secondary sodium systems

(7) Stop valves in primary sodium system

(8) Steam generator type, heat transfer pipe layout

The history of the basic Monju design is set forth in Table 1.1 stage by stage. After this, the fabrication design was implemented, and work was begun on the fabrication of equipment that was completely designed. This design work was continued to 1990.

Table 1.1 History of Conceptual Design of Monju

Preliminary Design—1968	Selection of crucial concepts	Preliminary design work was done, including the selection of reactor type, systems, equipment, and fuel, involving five Japanese equipment manufacturers, with a power output 300,000 kW in view. These selections were studied and evaluated, and the basic reactor specifications were determined.
Prototype Reactor, Primary Design—1969	Reactor core conceptual design	Based on the basic reactor specifications decided on, the system concepts, loop systems, and main equipment were subjected to comparative studies, and various parameter surveys were implemented.
Monju Preliminary Design—1970	Crucial equipment study	The FBR prototype was named "Monju." The main consideration in the primary design of Monju was the design of the major systems. Also implemented were the determination of the reactor structure nozzle positions, anti-earthquake design, and reactor core design.
Monju Secondary Design—1971	Overall plant design	With priority given to plant integration and safety, a harmonious plant design was developed.
Monju Tertiary Design—1972-1973	Site-conscious layout and safety design concepts solidified	Single-level pipeline approach (pulling primary cooling system main pipelines around high places), supplemental cooling system, reheat/non-reheat cycle study. Detailed piecemeal work, incorporating results of R&D and foreign information.
Adjusted Design—1973-1976	Licensable, reasonable design and conceptual design solidified	Proposal on other needed R&D items, design verifying plant viability.
Fabrication Preparation Design—1977-1979	R&D gains consolidated, design specifications re-examined, critical equipment items analyzed	Preliminary design from conceptual design to fabrication design (at this stage, the basic design of Monju was virtually complete).

The R&D on Monju can be divided into a first and a second stage, in terms of chronology, and as pertaining to design and construction.

The first stage continues up to construction start. In the early part of this stage, work began from relatively basic considerations, such as reactor physics, measurement and control, sodium technology, fuel, materials, and safety. Mockup tests were done for the main equipment, and other tests were conducted to obtain design data.

The second stage covered everything from construction start to the initial phase of operation. This involved R&D covering such things as prototyping, position and workability checks, maintenance and repair studies, for the purpose of detailed fabrication design.

(4) Licensing History

JAERI conducted a so-called check and review of the concrete Monju project, and the project was found to be acceptable in September, 1976. After this, official procedures began for the environmental inspections and construction work, with the guidance and cooperation of many involved parties, including local and administrative authorities.

In 1978 the environmental inspections were done, in 1979 other inspections were conducted according to the Natural Park Act, and all the environmental impact studies and investigations were completed in 1982. In the meantime, application was made for an atomic reactor installation permit in December, 1980. During the safety studies conducted by the authorities, concentrated deliberations were held by various groups of specialists responsible for the facility, the environment, and earthquake-resistance.

In May, 1982, local consent was obtained to go ahead with construction, and this was followed quickly by cabinet approval. After this, the stage-two investigations of the Atomic Energy Safety Committee were commenced, and hearings were held in July, 1982. In April, 1983, the Atomic Energy Commission and the Atomic Energy Safety Committee reported the results of their deliberations to the prime minister, and the latter granted the license for the nuclear reactor installation the following month. After that, the procedures for filing for various licenses pertaining to design and construction were filed for, and construction work got underway. It should be noted that, while Monju is a nuclear reactor that is in the R&D stage as set forth in the Regulatory Act Concerning Atomic Reactors, Etc., it nevertheless also comes under the provisions of the Electric Business Act.

(5) Site Survey and Construction History

In April, 1970, the Shiraki area in Tsuruga City, Fukui Prefecture, was selected as a candidate site for Monju, and Donen worked tirelessly to obtain consent to build, requesting the city to conduct a basic survey, under the guidance of the Science & Technology Agency (STA) and local authorities. Following the basic survey, in June, 1976, permission was granted by the prefecture to conduct preliminary surveys of the selected location. Weather observation facilities were installed on the survey site, and meteorological observations were made of wind direction and speed, temperature, and so forth, from October, 1976. Oceanic and geological surveys were also conducted, beginning at the survey site and covering a wide surrounding area.

In June, 1978, the results of more than 18 months of preliminary surveys were reported to Fukui Prefecture and, based on the results of the surveys, Donen drafted an environmental impact survey report which was submitted to the national and prefectural governments in August, 1978. Then, in February, 1979, Fukui Prefecture launched an investigation based on the Natural Park Act. This was completed in September, 1980.

Meanwhile, believing that the publicizing of responsible national opinion on the safety of Monju would be critical in obtaining local site approval, STA petitioned Fukui Prefecture for permission to conduct a safety investigation. This resulted in the prefecture consenting to the national authorities launching the safety investigation. Thereupon Donen petitioned the prime minister for an atomic reactor installation permit, and followed through on the aforesaid safety investigation.

After the atomic reactor installation permit was granted in May, 1983, and after all the permits based on the Natural Park Act pertaining to construction work were obtained, on 25 October 1985, notification of construction approval was received from Fukui Prefecture. Based on this, in the same month, actual construction work on Monju got underway. Thereafter, the large projects of doing the site work, construction, and equipment installation were conducted on schedule over a period of 5 ½ years.

During Monju's construction, many companies participated in the construction work, making the work on the site very congested. For this reason, it was most important to the progress of the work to coordinate everything with the overall project. The construction work moved ahead, with the construction processes being verified against the original construction plans using day orders, with minute construction process adjustments and the continued efforts of those doing the work.

Monju's equipment installation was completed in April, 1991, and trial operations began the following month.

2.2 Program

(1) Overview

In view of the fact that Monju is positioned as a national project, the development program involved the concerted efforts of public administrators, academia, and the private sector all over Japan.

In other words, Donen acted as the central developing agency, having primary responsibility, under the guidance and administration of the national government, and development work moved ahead with the cooperation of national and public research organizations and private industry. Furthermore, abundant opportunities were afforded to participating private industry from the perspective of raising the level of Japan's FBR technology. Thinking ahead to future technological transfer, ties with electricity businesses were strengthened from the construction phase on.

(2) Design and Licensing

Design work on Monju was begun in 1968 by Donen's FBR Development Headquarters.

Participation by private industry in executing specific design work under contract from Donen began with each atomic energy industry group during the period when the reactor type was being selected. As the design concepts began to be determined, what resulted was a form of design by joint division of labor involving the so-called atomic energy manufacturers, namely Sumitomo Heavy Industries, Ltd. (which subsequently withdrew), Toshiba Corporation, Hitachi Ltd., and Fuji Electric Co., Ltd. From about this time there were expectations that an organization would be set up to provide overall oversight and promote the comprehensive development of FBR technology. In 1977, the FBR Engineering Office was established, made up of the four companies just noted. This organization, later renamed FBR Engineering, Ltd., undertook the overall coordination and handling of all design work and technology. This entity handled the site construction start-up and technology, including construction work done by companies other than the big four, and became a major force in getting Monju completed.

All of the work involved in securing permits was handled by Donen.

(3) Research & Development

R&D on Monju was implemented concurrently with the design work which reflected the gains made with the experimental reactor Joyo, and there was a strong bias toward the design orientation. For example, the large sodium facilities were developed intensely by Donen's Oarai Engineering Center in direct linkage with the design work. Fuel R&D and

fuel design and fabrication were performed mainly by the Oarai Engineering Center and Donen's Tokai Office, respectively, while fast reactors of earlier vintage overseas were used for in-reactor irradiation work.

In developing the fast breeder reactor in Japan, we sought to do our own development work, based on our own technology, without dependence on foreign help. Hence our relations with foreign countries who had already developed FBRs was limited to the reciprocal exchange of technical information and to the use of existing reactor facilities. This determination to do our own development was grounded in the need to secure energy security and to raise the level of our industrial structure. But this determination also gets high marks in terms of its contribution to recent international cooperation concerning safety, and in terms of our taking part as a front-runner in FBR development.

(4) Fabrication and Construction

Donen has full responsibility for the construction of Monju. Nevertheless, the cooperation of the electrical companies was sought, to take advantage of their experience with light water reactors, and in view of transferring technology to the private sector in the future. For these reasons, Japan Atomic Energy Power Generation [*Nippon Genshiryoku Hatsuden*] Ltd. was given charge of those design areas had in common with light water reactors, and of the oversight of on-site construction.

Meanwhile, in the area of facilities, the four atomic energy manufacturers undertook responsibility for major systems in the reactor, but, as much as possible, many other companies were encouraged to participate. In particular, the site work and construction projects were divided into many segments and contracted out to many joint industrial bodies.

This approach was taken from the perspective of comprehensively enhancing domestic technology by giving many companies the valuable experience of working on Monju. And it is thought that this objective was very much achieved. On the other hand, however, monitoring and regulating processes, product quality, and safety in a huge project which involved more than 40 contracts, some 400 participating companies, and approximately 40 million man-hours of total labor, was a colossal job. Fortunately, everyone involved in this great project was fully aware of its significance, and proud to be participating. Thus the work was completed, without a single serious accident throughout the entire period of construction.

(5) Trial Operations

The trial operations conducted after all the equipment and facilities had been installed can be divided into two major categories. One was the comprehensive function tests done to verify the functionality of the equipment and systems prior to charging the reactor core with fuel. The other was performance tests done after the core was charged with fuel, for the purpose of verifying the overall performance of the plant.

Donen oversaw the comprehensive function tests, but they were actually conducted, with Donen's cooperation, by a joint headquarters organization set up by the manufacturers who had made the equipment. The performance tests, on the other hand, were to be conducted by Donen itself. This is being done with the participation of the Tokai Office in charge of fuel development, and the Oarai Engineering Center in charge of R&D.

In conducting these trial operations, half of the technicians (approximately 80 persons) were dispatched from electric power companies, and this also resulted in a definite promotion of technology transfer.

The performance tests were participated in by personnel sent from France, the United Kingdom, Germany, and the United States.

(6) Future Tests and Operational Framework

After the conclusion of the performance tests now being conducted, it is planned that Monju will be subjected to further tests as a prototype reactor for conducting R&D, and it will continue to be operated as an FBR power plant. The operational framework for these purposes will be based on the existing organization, but will be beefed up for the tests.

Meanwhile, the idea of employing Monju as an international exchange center for FBR technology is also being entertained.

Design

94FE0753B Tokyo GENSHIRYOKU KOGYO in Japanese Jun 94 pp 17-29

[FBIS Translated Text]

1. Design Policy[1]

1.1 System Design

The basic concepts selected for the fast breeder reactor (FBR) Monju are discussed below. The basic plant specifications are listed in Table 2.1.

Table 2.1 Basic Monju Plant Specifications

Reactor type	**Na-cooled loop type**
Heat output	714 MW
Electrical output	Approx. 280 MW
Fuel	Mixed Pu/U oxide fuel
Reactor core dimensions	
Equivalent diameter	Approx. 1.8 m
Height	Approx. 0.93
Plutonium enrichment (Pu_{fiss}%)	
Inside core/outside core	Approx. 16/approx. 21
Fuel charging volume	
Reactor core (U + Pu)	Approx. 5.9 t
Blanket (U)	Approx. 17.5 t
Fuel covering material	Stainless (SUS316 or equiv.)
Breeding ratio	Approx. 1.2
Reactor inlet/outlet temp.	Approx. 397°C/approx. 529°C
Secondary sodium system temp. (High-temp. side/low-temp. side)	Approx. 505°C/approx. 325°C
Reactor vessel dimensions (ht./diam.)	Approx. 18 m/approx. 7 m
Number of loops	3
Steam pressure (before main stop valve)	Approx. 127 kg/cm^2G
Steam temp. (before main stop valve)	Approx. 483°C
Fuel replacement interval	Approx. 6 months

(1) Reactor Type

There are two basic concepts for sodium-cooled FBRs, namely the loop type and the tank type. We conducted comparative studies of these two concepts in terms of structural characteristics, ease of operation and maintenance, safety, extrapolatability to practical reactors, and other R&D issues. The two subtypes have been constructed in about equal numbers overseas, each having their own advantages, and these will probably continue to coexist in the future. At any rate, for the reasons noted below, for the purposes of developing a prototype reactor in Japan, we decided to adopt the loop type.

- The loop type excels in ease of inspection, maintenance, and repair.
- With the loop type it would be possible to employ the technologies used in the high-speed experimental reactor "Joyo" and in light water reactors.
- At the time of the decision, the period of R&D required with the tank type tended to be longer.

(2) Fuel Replacement Method

During the preliminary design stage, we did comparative studies on the hot-cell plug manipulation and below-plug manipulation fuel replacement methods. The below-plug manipulation method requires a slightly more complex extraction mechanism, but the R&D period would be comparatively short, based on Joyo technology, making characteristic Japanese development possible. Priority was given to this point and we selected the below-plug manipulation method. This method, in turn, can be implemented with a number of different combinations, using single, double, or triple revolution, for example, and direct activation, fixed arm, or variable arm designs. In view of the reactor structure, size of the reactor core, and storage capacity, we decided on a single-revolution plug fixed arm approach.

(3) Fuel Aggregate Fuel Pins and Dimensions

The number of fuel pins (also called fuel elements) accommodated in the fuel aggregate is determined from the fuel pin outer diameter, pitch, ease of shaping, fuel replacement time, fuel exchangers, fuel insertion/extraction mechanism loading, and core thermal design, etc. Moreover, none of these can be determined independently. With Monju, carrying over from the Joyo experience, we determined the outer diameter and pitch of the fuel pins, and then settled on a fuel pin number of 169 as optimum in balancing the reactor structure with the fuel exchangers and fuel insertion/extraction mechanism.

The dimensions of the fuel aggregate are determined from the size of the fuel pins accommodated therein and the size of the materials. For Monju, we decided on a fuel pin length of approximately 2800 mm, a total length of approximately 700 mm for the top and bottom shields. In view of these dimensions, as well as the sizes of the handling heads and entrance nozzles, etc., we decided on a figure of approximately 4200 mm.

In designing the fuel spacers, design studies have been done concurrently on both grids and wire. Placing the main priority on pressure loss inside the reactor, we conducted water and sodium flow tests, together with rigorous analyses, and decided to go with wire.

(4) Temperature of Na Coolant and Steam Conditions

The steam conditions and sodium temperature in the primary and secondary cooling systems have a great impact on high-speed reactor heat efficiency, and on the size, reliability, and economy of equipment and pipelines. We implemented a survey of such parameters as burn rate, fuel pin pitch, gas plenum capacity, reactor core pressure loss, in-core flow velocity, pump power, core inlet/outlet temperature differential, and breeding ratio. We also considered the reduction of structural material creep and thermal shock, seeking an optimum situation.

As a result, we decided on sodium temperatures of 397°C at the reactor inlet, 529°C at the reactor outlet, and steam conditions of 487°C and 132 kg/cm^{2}G at the superheater outlet.

With regard to the steam conditions, we compared the reheat cycle at 127 kg/cm^{2}G and 483°C/483°C, and the non-reheat cycle at 127 kg/cm^{2}G and 483°C. We also conducted comprehensive studies on operation controllability, heat transient conditions, and economy, and finally adopted the non-reheat cycle.

(5) Position of Main Cooling System Pump

We conducted comparative studies on cold leg and hot leg installation. With cold leg installation, the pump shaft must be made long. However, it is used in a low-temperature condition, offers advantages in terms of heat transient conditions, and was used in Joyo. Hence we adopted the cold leg installation pump.

(6) Heat Generator Type and Heat Transfer Pipe Layout

Different countries use different steam generators of their own design. In Japan, however, giving thought to the integrity, reliability, maintainability, operability, and economy of future practical reactors, we decided against the modular type, and went with the unit type. Also, we decided to install the steam generator(s) and superheater(s) in separate locations, using heat transfer pipe materials suitable to each. We decided on a helical shape for the heat transfer pipeline. By doing so, we made it possible to use smaller, high-efficiency equipment. The helical shape also allows of a more economical plant design than do either the straight-pipe or hairpin configurations. Another consideration was the fact that Japanese manufacturers employ very sophisticated boiler, pipe fabrication, and welding techniques.

1.2 Safety Design

The safety design of Monju is based fundamentally on that of light water reactors, involving, as it does, the advantages of sodium cooling. Careful consideration was also given to the following safety issues.

(1) Reactivity Control

In a reactor core in which a uranium-plutonium oxide mixture fuel is used, just as in a light water reactor, the design is such as to exhibit peculiar negative reactivity characteristics due to the doppler effect, etc., throughout the entire operating range. Also, control of the reactivity within the reactor core is performed by control rods. There are two redundant reactor shutdown systems, namely a main reactor shutdown system that performs both a reactivity regulating function and an emergency reactor shutdown function, and a reserve reactor shutdown system which performs only an emergency reactor shutdown function. The reactor can be reduced to a low-temperature state with either of these systems alone, and is designed so that an uncritical state can be maintained.

(2) Elimination of Decay Heat After Reactor Shutdown

The elimination of decay heat after reactor shutdown is performed by three mutually independent systems, namely the primary cooling system equipment, some of the secondary cooling system equipment, and the auxiliary cooling equipment. Insofar as the elimination of decay heat is concerned, the primary and secondary cooling systems are each designed so that coolant can be circulated by means of a circulating pump driven by a pony motor connected to an emergency power system. Furthermore, in the unlikely event of a coolant leak from a pipe, the primary cooling system equipment is positioned above a stipulated reference height in order to insure a level of coolant liquid inside the reactor vessel sufficient to cool the reactor core. Equipment which of necessity must be located at lower levels is designed to be installed inside guard vessels.

(3) Design Considerations Regarding Sodium

The sodium used as the coolant excels in heat transferability and compatibility with the structural material, and has a high boiling point (880°C under normal pressure, approximately 980°C at core pressure), so that it can be used at high temperatures. On the down side, it is a chemically active substance, requiring design considerations such as the following.

(1) In equipment containing sodium, and having a liquid surface therein, an inert gas is introduced over the liquid surface in a structure wherewith sodium and air do not come in contact.

(2) Systems and equipment which are important to safety in circulating sodium are designed so that they will not lose their safety functions due to sodium solidification.

(3) Structures, systems, and equipment important to safety are designed so that they will not lose their safety functions due to the effects of sodium chemical reactions in the unlikely event that a sodium leak occurs.

(4) In rooms accommodating systems or equipment containing sodium, steel liners and the like are installed to prevent direct contact between sodium and concrete. These rooms are also designed so that water and other substances which react readily with sodium are removed as far away as possible.

(5) Systems and equipment which contain sodium are equipped, at suitable locations, with sodium leak detectors, and are thus designed so that sodium leaks can be detected early and the effects thereof suppressed.

(6) By placing the reactor vessel and the primary system pipelines and equipment inside the reactor containment vessel, even in the unlikely event of a mishap involving leakage of the primary coolant, the general public will be protected from radiation harm. Furthermore, rooms accommodating these systems and equipment are filled with nitrogen during reactor operation.

(7) The secondary cooling system equipment and pipelines are designed with sufficient excess structural strength against any reaction-induced pressure rise that might occur in the unlikely event of a water leak from the steam generator heat transfer pipes and subsequent sodium-water reaction, so that the effects of a sodium-water reaction can be suppressed.

(4) Storage of Radioactive Materials

As in light water reactors, the design features double protections to prevent the release of radioactive materials.

(1) The plutonium-uranium mixed oxide fuel pellets and fission products resulting from the radiation thereof are hermetically sealed inside fuel cladding pipes.

(2) In the event that, due to damage to a fuel cladding pipe, the radioactive material therein should leak out, such will be maintained within both the reactor coolant boundary and the reactor cover gas boundary.

(3) Against the possibility of the release of radioactive material outside the boundaries, due to the unlikely rupture of the reactor coolant boundary, etc., reactor containment facilities are provided to prevent the release of such material to the outside. The reactor containment facilities are made up of the reactor containment vessel and external shielding structures. This is a double containment structure, with a negative-pressure annulus provided between the two.

(5) Design Provisions Against Power Failure

As in light water reactors, multiple electrical power supply facilities are provided which exhibit a high degree of reliability, in a design that insures against their simultaneous breakdown. To this end, the external power supply system is connected by two circuits, while three separate diesel power generator units are provided to serve as on-site emergency power sources. The design insures against the loss of all power even for a short time.

(6) Physical Separation

As in light water reactors, for systems such as engineering safety facilities or safety protection systems which are important to safety and which feature redundancy and mutual independence, as necessary, the equipment, pipelines, and cables, etc., pertaining to each system, are either installed with adequate distances between them, or partitions are built between them, so that, in the unlikely event of a fire, etc., should one system become inoperable, the other system will not be affected thereby, and will not lose its safety functions.

2. Facility Layout

2.1 Overall Layout

The overall layout of the facilities within the site is diagrammed in Figure 2.1. The +42.8m elevation (hereinafter EL) at the center of the site was lowered to +21.0 m (in some places +31.0 m).

On the northeast side, on prepared ground at EL +42.8 m, are located, moving north to south, the solid waste storage facility, the maintenance and waste disposal building (hereinafter M/B), and the reactor auxiliary building (hereinafter A/B). On the south side, on prepared ground at EL +21.0 m, are located the turbine building, diesel building (hereinafter T/B and D/B, respectively), and central control building, etc.

The A/B which contains the reactor building (hereinafter R/B), is next door to the M/B, T/B, and D/B. Most of the access to all of these buildings is through the main entrance/exit of the D/B.

On the sea side are a breakwater wall and levee 0.8 km in length and a harbor for incoming shipments of heavy materials during construction and for shipping out spent fuel. The water for condenser cooling is taken from a place deep in the harbor and is discharged back into the harbor.

The site is accessed through a tunnel 0.9 km in length.

2.2 Overview of Reactor Construction, Etc.

(1) Re Layout

The A/B has a rectangular plan of approximately 115 x 100 m. The R/B (made up of the external shielding structures, steel containment vessel, and internal concrete structure(s)) is located roughly in the middle of the A/B in the configuration of the Japanese national flag.

This configuration contributes to earthquake safety and clarification of the structural plan.

The lifeform shielding walls inside the internal concrete structures have a hexagonal floor plan. This structure combines the usual shielding function with the facilitation of placing the reactor vessel and the three intermediate heat exchanges in closer proximity for greater efficiency.

The A/B is divided into a number of zones. The northern side is the fuel handling zone, the southern side is the steam generator zone, while the eastern and western sides are dedicated to control, electrical, and air-conditioning zones.

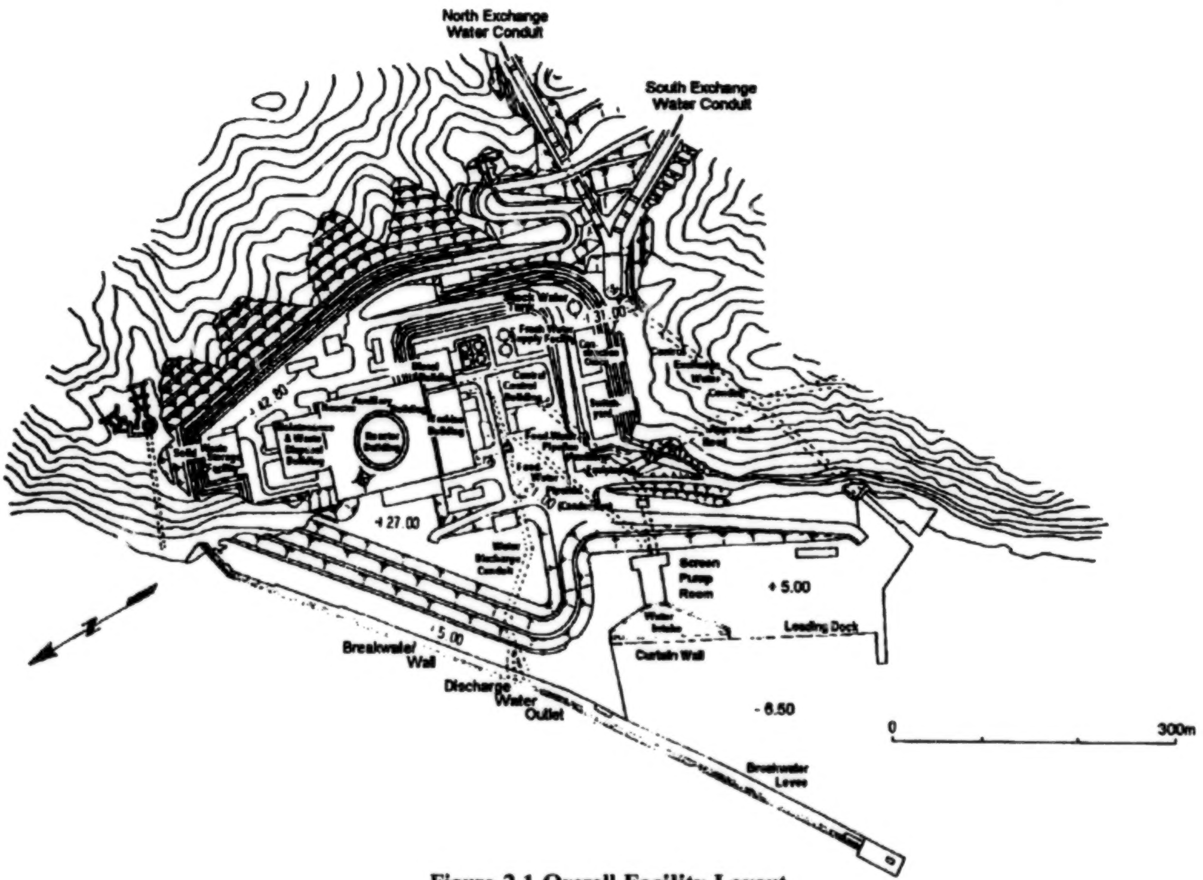


Figure 2.1 Overall Facility Layout

(2) Re Large Size of Buildings

The combined volume of Monju's R/B and A/B is roughly 630,000 cubic meters, which is larger than that found in a 1 million kW class light water reactor. The reasons why the buildings are larger are set forth below.

(1) The sodium used as the coolant must either be charged or drained when the reactor is started up or shut down, and the overflow tanks and dump tanks required for these operations take up much space.

(2) In order to mitigate the thermal stress generated by hot coolant, the pipelines are laid with many bends in them.

(3) Besides the ordinary air conditioning equipment, there must also be equipment for creating a nitrogen atmosphere inside the room housing the primary cooling system which is charged with radioactive sodium.

(4) Since sodium is chemically active, systems are provided to create an argon gas atmosphere above the liquid surface inside the sodium-containing equipment.

(5) Sodium gives off a lot of heat, so the air conditioning equipment must have a large cooling capacity.

(6) All fuel handling must be done by remote control, and the requisite equipment takes up much space.

(7) There are a great number of electrical boards for sodium pipe temperature maintenance and the like.

3. Reactor Core, Fuel

3.1 Reactor Core

The reactor core is made up of:

(1) the core fuel aggregates,

(2) the control rod aggregates,

(3) the blanket fuel aggregates,

(4) neutron shielding that surrounds (1) and (2).

The overall cross-section is hexagonal. The reactor fuel section is made up of two types of reactor fuel aggregate areas, having differing degrees of plutonium enrichment. By having the high-enrichment reactor fuel aggregate zones on the outside, a uniform two-region core is effected in which output averaging is implemented.

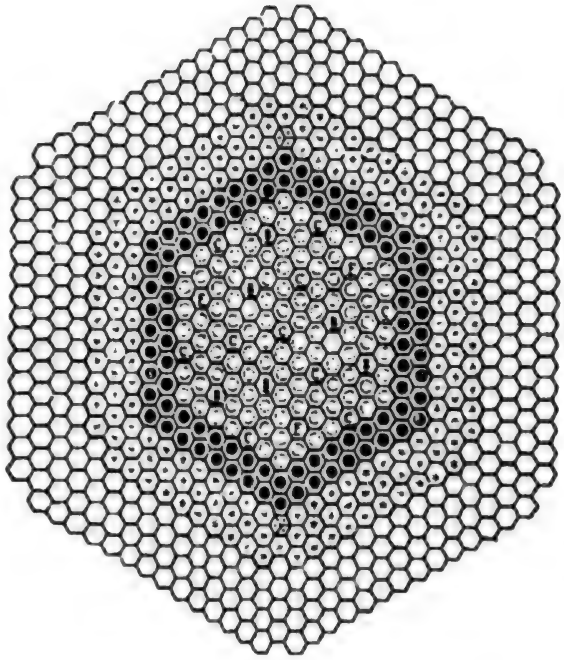
In the interest of evening out the reactor exit temperature, furthermore, there are 5 inner cores, 3 outer cores,

and 3 blanket regions, for a total of 11 reg.ons which are used for flow-volume distribution for a flow-volume design that matches the aggregate thermal outputs.

The reactor core fuel aggregates are configured with hermetically sealed core fuel that contains internal axial blanket fuel above and below, and is then provided, at top and bottom, with neutron shielding. Around the periphery of the core fuel aggregate region, blanket fuel aggregates are arranged in rings, so as to enhance the breeding ratio and reduce leakage of neutrons to the outside. Outside the blanket fuel aggregates, neutron shielding units are arranged, which both act as reflecting bodies, and reduce the neutron radiation penetrating to the outer structural equipment. The core equipment specifications are summarized in Table 2.2. The reactor core layout is diagrammed in Figure 2.2.

Table 2.2 Reactor Core Equipment Specifications

Core Fuel Region Shapes	
Number of regions	2
Effective height	Approx. 0.93 m
Equivalent diameter	Approx. 1.8 m
Blanket Thickness in Axial Dimension	
Upper	Approx. 0.3 m
Lower	Approx. 0.35 m
Equivalent Blanket Thickness in Radial Dimension	Approx. 0.3 m
Initial Fuel Charge Volume	
Core fuel regions—Plutonium and uranium	Approx. 5.9 t
Radial blanket—Uranium	Approx. 4.5 t
Axial blanket—Uranium	Approx. 13 t
Number of Core Fuel Aggregates	
Inner core	108 aggregates
Outer core	90 aggregates
Number of Blanket Fuel Aggregates	172 aggregates
Number of Control Rod Aggregates	19 aggregates
Number of Neutron Source Aggregates	2 aggregates
Number of Surveillance Aggregates	
Charged in neutron shielding region	8 aggregates
Charged in racks inside core	max. 8 aggregates
Average Core Fuel Take-Out Burn Rate	Approx. 80,000 MWd/t
Breeding Ratio	Approx. 1.2
Percentage Core Fuel Area Configuring	
Fuel	Approx. 33.5 vol%
Coolant	Approx. 40.0 vol%
Structural material	Approx. 24.5 vol%
Void	Approx. 2.0 vol%



Reactor Core Configurational Elements		Symbol	Quantity
Reactor Core Fuel Aggregates	Inner Core		108
	Outer Core		90
Blanket Fuel Aggregates			172
Control Rod Aggregates	Fine Adjustment Rods	F	3
	Coarse Adjustment Rods	C	10
	Backup Reactor Shut-Down Rods	B	6
Neutron Source Aggregates			2
Neutron Shielding			316
Surveillance Aggregates			8

Figure 2.2 Reactor Core Layout Diagram

3.2 Fuel

Two types of fuel are used, namely the core fuel aggregates and the blanket fuel aggregates. The core fuel aggregates, comprised by 169 fuel elements, are housed in hexagonal trumpet columns which have handling beds at the top and entrance nozzles at the bottom. The fuel elements are held in place within the trumpet columns by wire spacers so that they are arranged in the shape of an isosceles triangle. The core fuel elements have numerous pellets of plutonium-uranium mixed oxide toward the inside, but have blanketing uranium dioxide pellets, axially oriented, at the top and bottom, and are sealed inside cladding pipe.

The blanket fuel aggregates, which are almost identical on the outside to the core fuel aggregates, have 61 internal blanket fuel elements, held in the shape of an

isosceles triangle by wire spacers. The blanket fuel elements have numerous uranium dioxide pellets sealed inside cladding pipe, which are larger in diameter than are the core fuel elements.

Monju's fuel burn rate, initially, will be 55,000 MWd/t on the aggregate average, but ultimately will be 89,000 MWd/t on the aggregate average, reaching a burn rate of 94,000 MWd/t at aggregate maximum. With high-speed reactor fuel, the burn rate is set at higher levels than in a light water reactor, so fuel integrity must be maintained up to the high burn rate. To accomplish this, low-density, low-O/M pellets are used, together with a large plenum volume ratio, and high-performance stainless steel (SUS316 or equivalent) as the core material. More specifically, the fuel swearing which is produced by the buildup of solid and gaseous products of fission as the burn rate is advanced, is absorbed by the low-density pellets, which have a density that is roughly 85% of the ideal, and sufficiently large plenum volume is provided in the face of increased in-plenum pressure so that no excess is experienced.

In doing temperature analyses on the coolant, cladding pipe, and fuel pellets, in the course of establishing engineering safety coefficients, fabrication tolerances, physical value variations, and nuclear computational errors are all evaluated and the conservative computations made with adequate margins of error.

3 3 Control Rods

The degree of reactivity in the reactor is controlled by means of control rods. The control rods include 13 adjustment rods in the main reactor shutdown system, configured with three fine-adjustment rods and 10 coarse-adjustment rods, and six backup shutdown rods in the backup shutdown system.

Pellets of boron carbide formed with concentrated B^{10} are packed into stainless steel cladding pipes to form control rod elements, which in turn are bundled into supporting protective pipes. There are handling beds which make the connection with the control rod latch mechanisms in the drive mechanism, and shock absorbers or buffers which perform a buffering action at the end of an emergency reactor shutdown operation.

4. Reactor Structure and Cooling System Facilities

4.1 Reactor Structure

The reactor structure is made up of such members as a reactor vessel which incorporates the reactor core and the coolant flow channels, an inner-reactor structure which supports the core and provides for flow volume distribution to the various core components, and shielding plugs which shield off radiation and heat from the core and maintain an argon gas atmosphere above the liquid sodium surface. The overall structure is diagrammed in Figure 2.3 above.

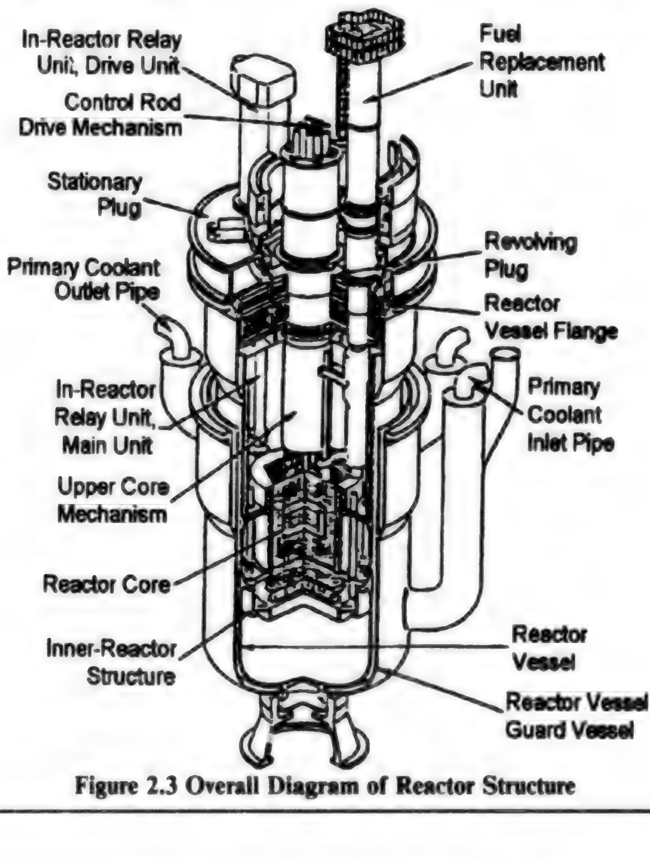


Figure 2.3 Overall Diagram of Reactor Structure

(1) Core Support

The reactor core is held in the center of the reactor vessel by the inner-reactor structure. The inner-reactor structure functions to prevent charging errors and to regulate flow volumes as respecting the fuel aggregates and other core configuring elements. This it does by means of fitted part dimensional combinations and the shape of its orifice(s). It is supported at the bottom of the reactor vessel by the reactor vessel.

(2) The reactor vessel is made of SUS304. It has an inner trunk diameter of 7.1 m, a trunk plate thickness of 50 mm, and an overall height of 17.8 m. It is an upright cylindrical vessel that is equipped with bottom mirror plates supported by an upper flange. The trunk is equipped with primary coolant inlet and outlet nozzles, as well as various small-diameter nozzles. At its bottom, the vessel is equipped with an inner-reactor structure attachment base. Also, vibration dampers are provided at the bottom of the reactor vessel which are structured to transmit horizontal earthquake loads to the building via the guard vessel.

The reactor vessel is designed with both earthquake safety and thermal stresses encountered during normal operation in mind. The upper trunk region of the reactor vessel is operated during rated operations at 529°C in the material creep temperature region. With consideration given to thermal shock conditions during transient times, thermal shielding plates are installed inside the upper trunk to reduce thermal shock, and a two-liquid level control system is adopted in the vicinity of the liquid surface.

A guard vessel is provided around the girth of the reactor vessel to maintain the liquid level required to cool the core even in the unlikely event of a sodium leak from the primary cooling system.

(3) Inner-Reactor Structure

The inner-reactor structure is a structural assembly that has an outer diameter of approximately 6.3 m and a height of approximately 6 m. It is configured with double-layer core support plates, support structures, and core tanks, etc. The inner-reactor structure has various functions, including (1) to hold the fuel aggregates and other core configurational elements in their proper positions, (2) to prevent these from being erroneously charged (loaded), and (3) to maintain the temperatures at the outlets of the fuel aggregates and other core elements as uniform as possible (flow-volume regulation function).

(4) Upper Reactor Mechanisms

The upper reactor mechanisms are made up of shielding plugs to shield off radiation and heat from

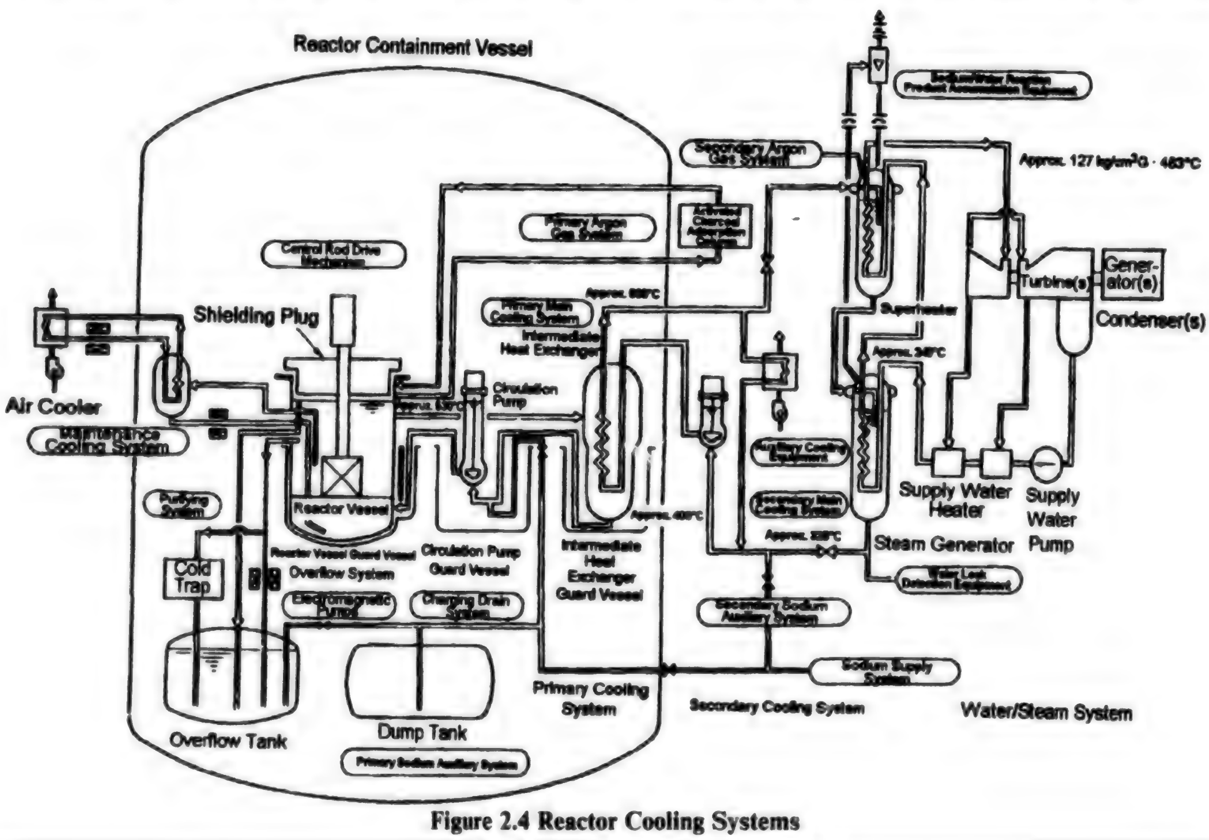


Figure 2.4 Reactor Cooling Systems

the core, a core-top mechanism comprising the control rod drive mechanism and core exit instruments and equipment, and a pantograph opening/closing fuel replacement unit that is loaded on at fuel replacement time.

The shielding plugs—for which a simple revolving plug system is adopted—consist of a stationary plug and a revolving plug. The stationary plug has a diameter of approximately 9.5 m and the revolving plug a diameter of approximately 5.9 m.

The core-top mechanism is positioned inside high-temperature sodium, so ample thought is given to its suitability in terms of thermal transient conditions, and temperature-fluctuation-induced high-cycle thermal loading (by employing protective structures made of alloys with super-high heat resistance, etc.).

4.2 Reactor Cooling System Equipment

The reactor cooling system equipment takes the heat generated in the reactor core and leads it to the turbines and generators. The reactor cooling system is made up of a three-system primary main cooling system, a secondary main cooling system, and a one-system main steam system. The cooling systems are diagrammed in Figure 2.4.

(1) Primary Main Cooling System Equipment

In the primary main cooling system equipment we have a circulation pump, pipelines, stop valve(s), and electromagnetic flowmeters, etc.

The primary main cooling system's circulation pump is an upright mechanical free-liquid-surface centrifugal pump which produces a flow volume of approximately 5.1×10^6 kg/h and a lift of approximately 90 mNa. This pump is installed in the pipeline on the low-temperature side of the system. This pump must maintain a constant temperature differential between the reactor inlet and outlet during normal operation. To accomplish this, the circulation flow volume is controlled within a range of approximately 50 to 100% by variable frequency power supply equipment that is linked to the reactor output. During a reactor scram, a certain quantity of coolant must be circulated in the reactor core. For this purpose, a small motor (called a pony motor) is connected to an emergency power supply so that the prescribed flow volume can be secured.

The pipeline which configures the circulation loop in the primary main cooling system is installed horizontally, at a higher position than the reactor core, so that the liquid level necessary to cool the reactor core can be maintained even in the event of a sodium leak from the cooling system pipelines. Also, guard vessels are provided for the primary main circulation pump that is positioned at roughly the same height as the reactor, the intermediate heat exchanger, and the pipelines connected to these units.

(2) Intermediate Heat Exchanger

The intermediate heat exchanger is an upright, non-liquid-surface, parallel counterflow type of unit that transfers heat generated in the reactor core from the primary main cooling system to the secondary main cooling system. This heat exchanger has a heat-exchanging capacity of approximately 238 MW. The outer trunk diameter is approximately 3 m, the overall height 13 m. The heat transfer pipes are made of SUS304. To achieve compactness and high performance, the array pitch is contracted, and the wall thickness is reduced but not at the expense of adequate strength. Flow-streamlining mechanisms are installed at various places in the exchanger to make the flow distribution uniform.

(3) Secondary Main Cooling System

The secondary main cooling system's circulation pump is of the same type as that of the primary main cooling system. It develops a flow volume of approximately 3.7×10^6 kg/h and has a lift of approximately 50 mNa. It also is installed on the low-temperature side of the system. The pump circulation flow volume can be controlled within a range of approximately 50% to 100% by means of a static variable frequency unit.

The removal of the decay heat during reactor shutdown is accomplished by air coolers in auxiliary cooling equipment installed in parallel with the steam generator(s) in the secondary main cooling system. The heat is ultimately released into the atmosphere. Each of these air coolers has a heat removal rating of approximately 15 MW. This rate of heat removal from the reactor core during reactor shutdown is possible with only one system.

(4) Steam Generator

For the steam generator, the helical-coil, through-flow type of unit—in which the evaporator and superheater are separated—was adopted in the interest of small size and in view of the material selected for the heat transfer pipes.

The evaporator has a heat transfer volume of approximately 190 MW, an overall height of approximately 13 m, and a trunk diameter of approximately 3 m. The evaporator has built into it 140 Cr-Mo steel heat transfer pipes which excel in resistance to stress, corrosion, and cracks. These pipes each have an outer diameter of approximately 31.8 mm and a wall thickness of 3.8 mm. During standard rated operation, the unit generates steam (at approximately 369°C) having a superheated degree of 30°C or higher.

The superheater has a heat transfer capacity of approximately 50 MW, an overall height of approximately 10 m, and a trunk diameter of approximately 3 m. It has built into it approximately 150

heat transfer pipes made of austenite-based stainless steel which excels in resistance to corrosion against steam. These transfer pipes have an outer diameter of approximately 31.8 mm and a wall thickness of 3.5 mm. The superheated steam flowing in from the evaporator is superheated further to approximately 490°C/132 kg/cm^{2}G and sent to the turbine(s).

The structure of the evaporators which make up the steam generator is depicted in Figure 2.5. The structure of the superheater is almost the same.

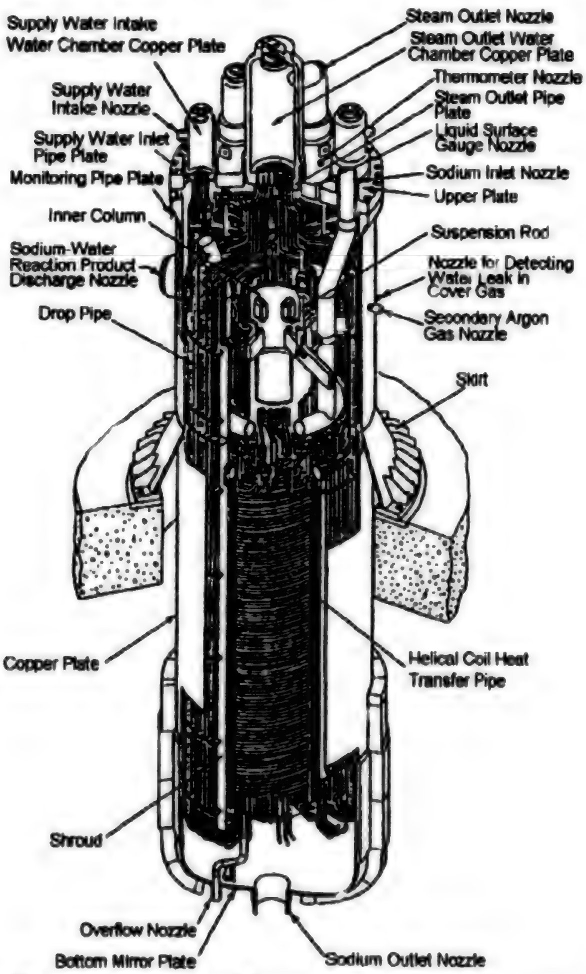


Figure 2.5 Structure of Evaporator for Steam Generator

Smaller size was achieved in the steam generator by adopting the helical coil structure. The number of welds between pipe and pipe plate were also reduced, and pipe plate thermal stress was reduced. The structure also permits pull-out removal by connecting the pipe bundle to the upper trunk section.

5. Water, Steam System Equipment

The water and steam system equipment includes the steam turbine(s) and auxiliary equipment, water supply system equipment, main steam system equipment, and condensed water system equipment. The supply water, supplied by two turbine drive water supply pumps, is heated in the steam generator (evaporator and superheater), made into superheated steam, then passes through the main steam check valve and steam regulating valve, and into the high-pressure turbine, two low-pressure turbines. These turbines drive generators that are rated to develop approximately 280,000 kW of power.

The steam generators are sodium-heating separation pass-through types. Therefore, in designing the water and steam systems, the following things are done.

(1) During Plant Start-Up

- A boiler is provided as an auxiliary steam source to heat the supply water so that the water passing through the steam generator is not cooled below the solidifying temperature of sodium.
- In order to prevent flow instability phenomena in the evaporator, at the point in time when steam is produced, the supply water flow volume is not allowed to fall below a certain level.
- To preserve the integrity of the heat transfer pipes in the superheater, until superheated steam is provided, the steam bypasses the superheater through a start-up bypass system.
- At start-up, when the turbine extraction steam cannot be used, the steam exiting the steam generator is used to heat the supply water.

(2) During Output Operation

- To regulate the quality of the supply water, it is subjected to volatile chemical treatment using ammonia and hydrazine so as to preserve the integrity of the steam generator heat transfer pipes.
- In order to handle rapid turbine load reductions, a high-speed operating turbine bypass system is provided to secure the stable operation of the water and steam system.
- In a reactor scram, the steam is blown, and the pressure and temperature inside of the steam generator are reduced to prevent a temperature rise in the low-temperature section of the steam generator.
- In the unlikely event of a water leak from a steam generator heat transfer pipe, the water supply is interrupted, and a release valve is activated to release the water inside the steam generator to the outside, and thereby prevent the mishap from becoming more serious.

The steam turbines are tandem-compound, three-cylinder, four-flow-exhaust, non-reheating condensed water turbines. The temperature at the turbine inlet under steam conditions at rated output is approximately 483°C, the pressure is 127 kg/cm^{2}G, and the exhaust

vacuum is approximately 722 mm Hg. Of the steam turbines, the high-pressure turbine must withstand high pressures and temperatures, so it is designed along the lines of a thermo-power turbine. The low-pressure turbines are also non-reheating types, and are designed along the lines of light water reactor turbines, with the focus on drain measures in connection with corrosion prevention.

6. Fuel Handling System Equipment

6.1 Equipment Summary

The fuel handling system equipment must safely and securely handle and store the elements which make up the reactor core—including the core fuel aggregates, blanket fuel aggregates, control rod aggregates, and neutron shielding units—from the time they are shipped to the plant site until the time that the used core elements are shipped back out of the plant site. Most of the fuel handling system equipment is positioned on the north side of the reactor auxiliary building, with the transfer of core elements back and forth between the facilities being done by the fuel input/output equipment. This equipment is arranged in series along the rails on which the fuel inserter/extractor runs. These pieces of equipment are operated by remote control. The fuel handling routes are diagrammed in Figure 2.6.

The fuel is replaced once every 6 months or so, when the reactor is shut down. During fuel replacement, the equipment conveyor entrance to the reactor containment vessel is opened, and approximately 20% of the total reactor fuel is replaced using the fuel insertion/extraction equipment.

During reactor operation, the new core configuring elements that are received and stored in new fuel storage racks are transferred to the out-of-reactor fuel storage tank in preparation for the next fuel exchange. At the out-of-reactor fuel storage tank, spent core elements which have been in storage cooling for approximately 1.5 years to reduce their radioactivity are washed, put into sealed canisters, and stored in storage racks in the underwater fuel storage facility. The transporting of these new fuel and spent core elements is performed by means of the fuel insertion/extraction equipment, the underground trolley, and the underwater trolley.

6.2 Fuel Replacement Equipment

The fuel replacement equipment is used to exchange and transport reactor core configurational elements inside the reactor, and consists of the fuel replacement unit and the in-reactor relay unit.

During fuel replacement, the fuel replacement unit is put into place over the revolving plug, and, linked to the turning of the revolving plug, it takes reactor core elements and transports them between the reactor core and the in-reactor relay mechanism. It is made up of the main fuel replacement unit and a hold-down arm. The main unit, made up of a gripper, pantograph, and drive

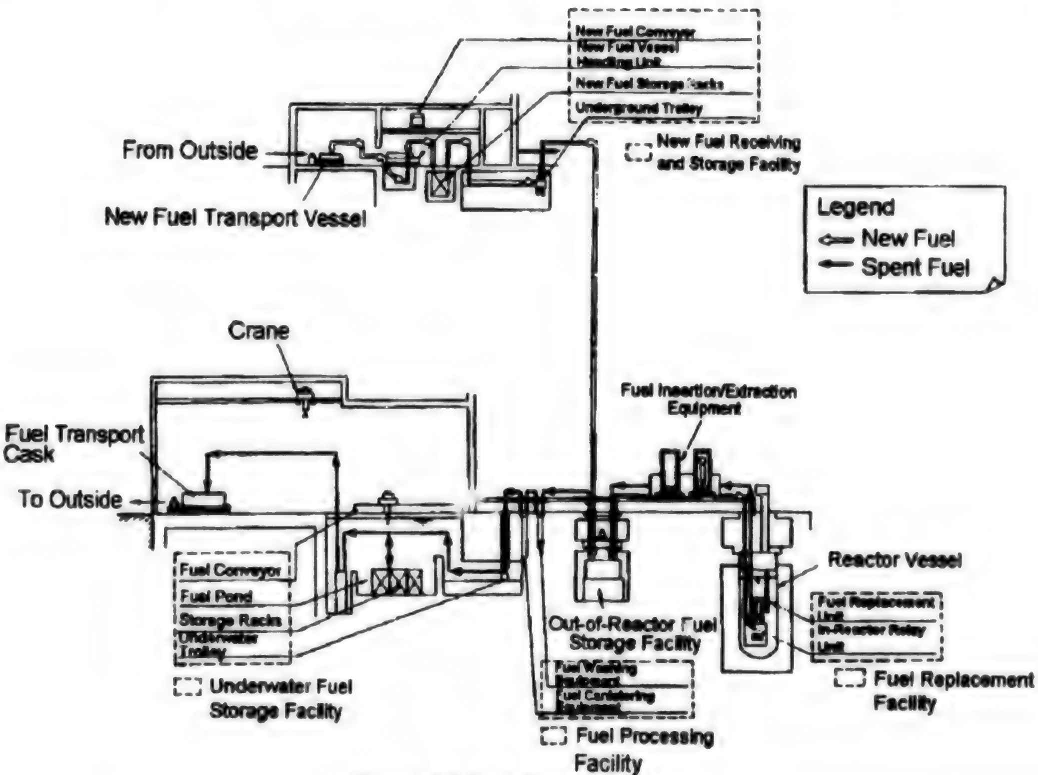


Figure 2.6 Fuel Handling Routes

mechanism, opens and closes the pantograph, grips and releases the fuel, and raises and lowers the main unit. The hold-down arm guides the main unit and provides anti-earthquake support. This arm is always inside the reactor, so its structure has a horseshoe-shaped cross-section to provide earthquake-resistant rigidity and thermal stress relaxation. Thermal shielding plates are attached to the outside surface of the arm.

The in-reactor relay unit is installed on top of the stationary plug during fuel replacement. Its function is to transfer core elements contained in fuel transport pots between the fuel insertion/extraction equipment and the in-reactor relay unit revolving rack.

6.3 Fuel Insertion/Extraction Equipment

The fuel insertion/extraction equipment performs a number of functions. During fuel exchange, it transports core elements between the interior of the reactor and the fuel storage facilities outside of the reactor. During reactor operation it transports new core elements to the fuel storage facilities outside the reactor and removes used core elements from the fuel storage facilities outside the reactor and conveys them to the fuel processing facility and the underwater fuel storage facility.

The fuel insertion/extraction equipment consists of a fuel insertion-extraction unit A which handles core elements that have sodium adhering to them, a fuel insertion-extraction unit B which handles used core elements after the sodium has been washed off, a cooling unit which cools the used core elements during transport, and a trolley which carries these units. Out of consideration for the fuel handling equipment positions and core element transport, the trolley runs on a straight track, and grips the track rail girders to achieve high earthquake resistance.

6.4 Out-of-Reactor Fuel Storage Facility

The out-of-reactor fuel storage facility temporarily stores new core elements that are to be loaded into the reactor core and used core elements taken from the reactor core, in sodium. The facility consists of an out-of-reactor fuel storage tank and auxiliary equipment. The storage tank has the capacity to store approximately 250 core elements. It is equipped with (a) revolving rack(s) having six concentric rings of storage positions. The insertion and extraction of the core elements is performed by turning the revolving rack(s) with a drive unit.

6.5 Fuel Inspection Equipment

The fuel inspection equipment performs inspections to check for damage to fuel aggregates removed outside the reactor.

6.6 Fuel Processing Facility

The fuel processing facility consists of the fuel washing equipment and the fuel canistering equipment.

The fuel washing equipment washes the adhering sodium from the used core elements that are removed, after they have been temporarily stored and cooled in the out-of-reactor fuel storage facility.

The fuel canistering equipment works by remote control to canister the used core elements (fuel aggregates and control rod aggregates) after the sodium has been washed off.

6.7 Underwater Fuel Storage Facility

The underwater fuel storage facility stores used core elements in a pool of water. It has a storage capacity of approximately 1400 units.

6.8 New Fuel Reception & Storage Facility

The new fuel reception and storage facility takes in new core elements, stores them temporarily, and transfers them to the fuel insertion/extraction equipment. It can store approximately 50 new core elements in air.

7. Measurement & Control Equipment

The configuration of Monju's measurement and control equipment is basically the same as in a light water reactor. However, since Monju uses sodium as the coolant, it reflects years of R&D gains in its sodium instrumentation. We will now discuss those aspects of Monju's measurement and control equipment that are characteristic.

7.1 Plant Control System

When we compare Monju's cooling system to that of a light water reactor, we see that:

(1) the reactor entrance/exit temperature differential is greater,

(2) it is a system that has large thermal capacity and heat transport time, being configured with a primary and a secondary system,

(3) the main steam system uses superheated steam turbines, for which reason the main steam temperature and pressure must be maintained constant.

For these reasons, in controlling the plant, the reactor output and main cooling system flow volumes are regulated by plant output commands, with the turbine generator output tracking behind. The plant is controlled automatically within a range of 40% to 100% of rated output by target value settings. The plant is designed so that, within this automatic control range, it can handle:

(1) +/-5%/minute ramped output fluctuations,

(2) +/-10% stepped output fluctuations,

(3) turbine load reductions of 50% or less by activating the turbine bypass valve(s).

7.2 Central Control Room, Control System

The operational monitoring and control necessary to the operation of the main plant systems are conducted from the central control room. A central monitoring panel is situated in the middle of the central control room, making it possible to monitor and control the plant while seated. Behind this is a bench-type central control panel for controlling the plant during normal and emergency operations. In designing this central control room, the fundamental policy was to reduce operator load and prevent errors in judgment and execution. Extensive use is made of computers and CRT screens. In designing the instruments and operating switches on the central monitoring panel and central control panel, a wooden mockup was employed, and ergonomic studies were implemented. This resulted in a rational panel layout.

7.3 Fuel Integrity Monitoring Equipment

The damaged-fuel detection system detects damages in the unlikely event that a fuel pin ruptures, and determines the location of the damage. It is made up of (a) cover gas method detection unit(s), delayed-generation neutron method detection unit(s), and tagging-method position detection unit(s). The cover gas method detection unit detects FP gas released in the reactor cover gas when a fuel pin ruptures. It is highly sensitive. The delayed-generation neutron method detection unit detects delayed-generation neutrons from the FP that is released in the sodium when a fuel pin ruptures. It is very responsive. The tagging-method detection unit is a system in which a mixture of Krypton and Xenon gas is sealed in a predetermined composition into each fuel pin beforehand. In the unlikely event of a fuel pin rupture, the gas released thereby is analyzed and the ruptured pin thereby identified.

7.4 Sodium Leak Monitoring System

Sodium leak monitors are positioned throughout the system, wherever necessary. In the unlikely event that sodium leaks from a pipe or something, sodium aerosol is produced. This is detected by sampling nozzles which are attached to the periphery of the pipelines and so forth and continuously take samples. Both sodium ionization detectors and pressure-differential detectors are used. The signals from these detectors are processed with high detection sensitivity to insure early detection of a leak.

7.5 Water Leak Monitoring System

Heat is exchanged between the sodium and the water/steam via heat transfer pipe walls in the steam generator. In the unlikely event of a water or steam leak from one of these heat transfer pipes, the sodium would react with the water to produce hydrogen. The water leak monitoring system measures changes in hydrogen concentration and thereby detects water leaks. Monju is equipped with two types of system, namely one which samples the coolant sodium (from the superheater outlet pipeline, the evaporator outlet pipeline, and the secondary main circulation pump inlet pipeline, respectively), and another which samples the cover gas inside the steam generator. By measuring the pressure of hydrogen which passes through a nickel film in hydrogen meters (which separate hydrogen with a nickel film) inside the detection equipment, water leaks are detected.

8. Electrical Equipment

The electrical power generated by Monju is transmitted over a transmission system that includes two 275-kV transmission lines. Each of these lines is capable of transmitting all of the power generated at the plant. Thus the plant can continue operating at full output even if one line goes down.

The generators are completely sealed three-phase synchronized generators which use hydrogen gas as the cooling medium. The cooling system used for the rotors employs direct hydrogen gas cooling in a diagonal flow. Direct water cooling is used for the stators, with pure water being passed through internally. The thyristor excitation method is employed, with the excitation power supplied from excitation transformers connected to phase-separated mains.

9. Security Measures, Nuclear Material Protections

Incorporated into the design of Monju are security measures, nuclear material protections, and other measures to prevent nuclear proliferation. These measures are incorporated because we are handling the highly sensitive substance plutonium, in the form of plutonium-uranium mixed oxide fuel (hereinafter "MOX fuel").

With the Japan-IAEA Security Measures Agreement, the Nuclear Material Protection Treaty, and the New Japan-United States Nuclear Cooperation Agreement, specific requirements pertaining to nuclear nonproliferation are spelled out. These requirements have been incorporated into Monju from the design and construction phases, making it a facility that is on the cutting edge of technology.

9.1 Security Measures System

The design of Monju's security measures system was undertaken early, resulting in July, 1991, in an agreement between the Japanese government and the IAEA, after much deliberation, on a facility memorandum.

Besides an annual inventory verification, Monju inspections include a monthly interim inventory verification due to demands for timeliness. For this reason, Monju's security measures system employs many containment systems and monitoring systems, based on the principle of nuclear material weight monitoring. Since Monju is a sodium-cooled reactor, it is not possible to directly inspect the fuel inside the reactor (in the unapproachable sector), so the method employed is to inspect the loading of new fuel into the reactor, and to inspect the spent fuel taken from the reactor, using radioactivity monitors (entrance gate monitor, fuel insertion/extraction unit

monitor, exit gate monitor), thereby streamlining the inspection process and avoiding the human approach problems.

Thus many monitoring devices are employed which are automated. This also reduces inspection workload.

In April, 1991, the blanket fuel was loaded into the Monju reactor, and the first inspection was carried out the following June. Since the introduction of MOX fuel in July, 1992, periodic inspections have been performed.

9.2 Nuclear Material Protection System

The Monju site is surrounded on three sides by mountains, with access thereto by tunnel. Thus the site is very unusual topographically. Also, since the site is situated on the tip of a peninsula, it is necessary to take measures to effect nuclear material protections which give due consideration to the response time required by the proper authorities.

Furthermore, Monju is subject to strict nuclear material protection conditions in view of the types and quantities of nuclear material handled.

Based on these conditions, in Monju's nuclear material protection system, the plant is divided into sectors (most of which are buildings) where rigorous security measures are needed. The security systems are designed and implemented on the basis of experience in implementing nuclear material protections under Donen (Power Reactor and Nuclear Fuel Development Corporation), and incorporate the latest technology and security know-how.

Since the codification of legal regulations for protecting nuclear material in 1989, Donen has also been operating a nuclear material protection system in a rigorous program, in view of the fact that this is a new facility being operated for the first time.

10. Simulator

The Monju simulator (MARS = Monju Advanced Reactor Simulator), is the first full-scope simulator in the world which simulates a fast breeder reactor power plant. In designing this simulator, the experience in designing the hardware and software for the simulator used for the FBR Joyo was used to good advantage. The operating training used with that reactor was also found beneficial. This simulator was installed adjacent to the plant so as to perform three functions:

(1) to evaluate plant operation and control performance and operation procedures,

(2) to provide training both in normal and emergency operations,

(3) to efficiently support the development of future operating support systems and emergency diagnostic systems in the interest of effecting further sophistication.

The simulator can handle approximately 280 malfunction cases.

Research and Development Toward Monju

94FE0753C Tokyo GENSHIRYOKU KOGYO in Japanese Jun 94 pp 30-41

[FBIS Translated Text]

1. Overview

At Donen (Power Reactor and Nuclear Fuel Development Corporation), with a view to the development, using Japanese-developed technology, of the experimental reactor Joyo and the prototype reactor Monju, R&D is being pursued in the fields of reactor physics, shielding, fuel, safety, sodium technology, equipment, structural materials, and measurement and control, working jointly with national institutions (JAERI, universities, national and public research institutes, electric power companies, and manufacturers). Donen has also been working aggressively to promote development through international cooperation.

Intent on implementing R&D in a focused and efficient manner, in March, 1970, the Oarai Engineering Center was opened, complemented with researchers and a high concentration of various large experimentation facilities and equipment. The initial R&D gains were reflected in Joyo, and, after Joyo went critical (April, 1977), in the design, licensing, and construction of Monju (together with what was learned in the operation of Joyo).

In developing Monju, particular weight was given to problems involving the reactor core and fuel, equipment and structure, and safety.

Concerning the reactor core and fuel, the issues were how to enhance reactor output and effect high burn rates in the fuel. In joint Japan-UK research (the MOZART project), the propriety of the reactor core design was confirmed, and mixed oxide fuel (MOX) test units fabricated at the Tokai installation were irradiated in foreign reactors and in Joyo to determine the irradiation characteristics.

Concerning equipment and structures, the Monju structural design was rationalized, working from the Prototype Reactor High-Temperature Structural Design Standards, and the steam generator system needed for a power plant was developed also. Tests and analyses were done using large sodium test facilities. Not only were the engineering problems associated with scaling up the equipment addressed, but measurement and control methods were also developed.

Concerning safety, experimental and analytical studies were done to determine various phenomena pertaining to the reactor core, the decay heat removal systems, and the containment system. Safety evaluation procedures were developed through experimental research at the

Oarai Engineering Center and through joint Japan-France and Japan-United States research. Research was also done on sodium-water reactions. These findings were reflected in Monju's safety design and safety examination.

2. Reactor Physics

2.1 Reactor Core Nuclear Design Computation R&D

Reactor core nuclear design is positioned most upstream in atomic reactor design work. For this reason, such R&D was begun early. The MOZART (Monju Zebra Assembly Reactor Test) project, discussed below, was launched more than 20 years ago (September, 1971 - March, 1973).

As an example of how the precision of nuclear computations can affect design, there is a case in which a 1% error in the effective breeding rate of an FBR resulted in miscalculating the number of fuel aggregates needed for core criticality by the margin of 15. Accordingly, for Monju, simulated criticality tests were done in advance, using a criticality testing system. It was necessary to evaluate the precision of the computational procedures (including modeling procedures) for the nuclear constants used in design calculations. In the Joyo case, simulated criticality tests were conducted using the FBR criticality testing system (FCA = fast critical assembly) of JAERI. Monju has a larger reactor core than Joyo, however, and it was not possible to implement simulated criticality tests of the entire core. Thereupon, these tests were done using ZEBRA, a criticality testing system of the UK (MOZART project).

In the MOZART project, not only were criticality tests conducted, but also almost all of the tests on major nuclear characteristics—including output distribution, sodium void reactivity values, sample reactivity values, density coefficients, and control rod reactivity values—were done. In order to supplement the MOZART project findings, analyses of criticality tests and the like were also conducted using FCA and U.S. criticality testing systems ZPR and ZPPR.

Through all of these criticality test analyses, the reliability of the design was confirmed.

2.2 Shielding Design Computation R&D

Classic procedures such as the removal and dispersion method and point kernel method were used in designing the shielding for Joyo and in the conceptual design of the Monju shielding.

Shielding measurements were made in the Joyo MK-I reactor core between 1977 and 1979. In these evaluations, use was made of the two-dimensional S_N transport computation code DOT 3.5, which had never been employed in Japan in large-scale shielding analysis. The procedures used in these Joyo MK-I shielding test analyses (the computation flow is diagrammed in Figure 3.1) were verified as to their validity in Joyo. This represented a major advance over previous analysis, and was used with almost no modification in the Monju design work.

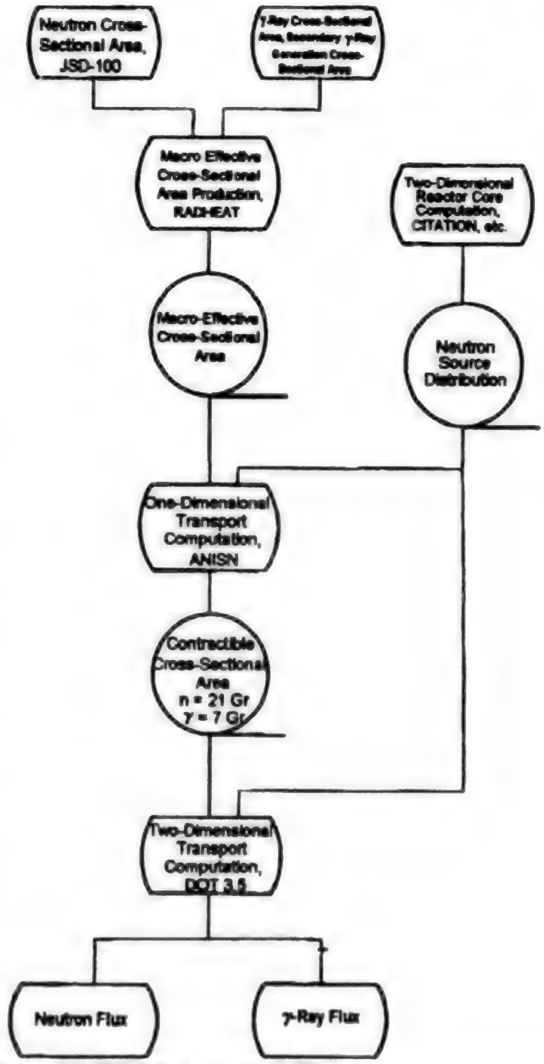


Figure 3.1 Shielding Analysis Flow

The techniques used in current large-scale reactor shielding computations are fundamentally the same as those used in the shielding computations for Monju. R&D was also conducted for the Monju shielding design. This included, in addition to the work done with the Joyo MK-I reactor core, shielding measurements made with the Joyo MK-II reactor core, streaming experiments with the Yayoi reactor, and shielding measurements with the FFTF reactor in the United States. The data obtained were used in the design.

2.3 Development of Other Nuclear Computation Codes

FPGS was developed as a computation code for decay heat and radiation quantities, etc. Unlike ORIGEN,

which is also used in these calculations, what is characteristic of the FPGS code is that it facilitates the use of multiple groups of contractible cross-sectional areas in any spectrum, rather than just one group of fixed cross-sectional areas contractible in a specific spectrum. Thus FPGS code permits computations of greater precision.

For Monju's reactor core control code, the 3D Hex-Z (that which takes a hexagonal mesh in the radial direction) system dispersion computation code which uses a modified coarse mesh method was newly developed. While this technique is a Hex-Z system, it promises the same degree of computational precision as the Tri-Z (that which divides the aforesaid hexagon into 6 segments, and takes a hexagonal mesh in the radial direction) system.

Also developed was HIBEACON. This is not a nuclear computation code, but rather a reactor core curve design code which is necessary in designing a reactor core. This was subsequently modified slightly, and is still used in large reactor core design.

3. Fuel, Materials

3.1 Fuel Development

Compared to the fuel pin average burn rate of 75,000 MWd/t in Joyo, the targeted burn rate for the Monju's fuel was approximately 100,000 MWd/t (with a pellet peak burn rate of 130,000 MWd/t), which is high. The performance and behavior of such fuel could only be ascertained inside an actual reactor. This being so, irradiation tests and other R&D work was begun, mainly with the irradiation testing facilities and Joyo of the Oarai Engineering Center.

Initially, researchers conducted burning acceleration irradiation tests at DFR and Rapsodie, irradiation tests, with measurement lines added, using the Joyo MK-II irradiation reactor core, high-line output irradiation tests, and various other irradiation tests under a wide range of irradiation conditions and fuel specifications. Armed with these irradiation data, the fuel temperature evaluation precision was enhanced, and detailed behavioral analyses were conducted on FP gas release rates, fuel-cladding chemical interaction (FCCI), and fuel element transformation. Based on these findings, it was confirmed that the physical property evaluations and design techniques used in designing the fuel were sufficiently conservative. The data and knowledge obtained were also reflected in the development and verification of the fuel behavior analysis code CEDAR.

Aggregate irradiation tests were conducted, both at Joyo and FFTF, to verify the integrity of the fuel specified for the actual reactor, a pellet peak burn rate of 147,000 MWd/t (higher than the targeted burn rate for Monju) was achieved, and it was verified that this fuel is safe and will not rupture.

3.2 Reactor Core Material Development

Monju's fuel cladding reaches a peak high-speed neutron irradiation dosage of approximately 120 dpa, so it was important to develop cladding pipe that could withstand use under such irradiation. Number 316 stainless steel, excelling in high-temperature strength, was selected as the cladding pipe material, but it was also necessary to inhibit the void swelling that is caused by high-speed neutron irradiation. To this end, numerous prototype evaluation tests were conducted, and it was demonstrated that the introduction of cold-process-induced transitions and the addition of minute quantities of elements to diffuse microscopic deposits were effective in improving swelling resistance. After considering weldability and ease of fabrication, optimization was sought by compound additions of minute elements (P, B, TI, Nb) to 316 stainless steel, and by cold processing.

The end result was success in developing a material that, for swelling resistance and creep strength, was at the

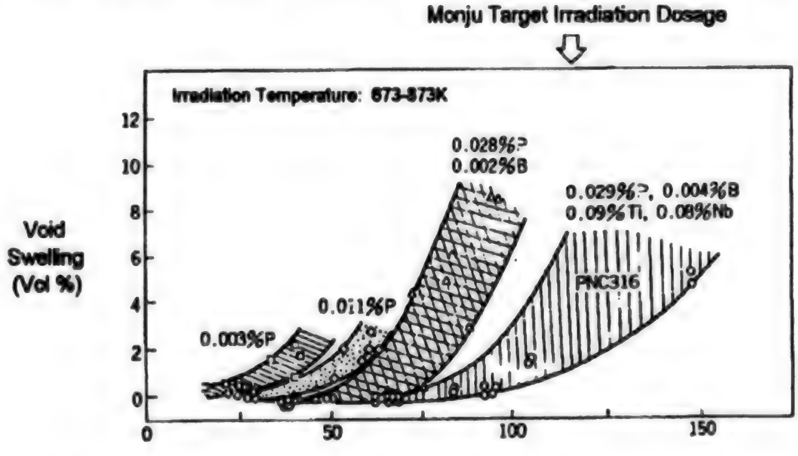


Figure 3.2 Improvement of Swelling Resistance in 316 Stainless Steel

highest level of any material in its class in the world. This was named PNC316 steel (also called SUS316 equivalent steel). The improvement in swelling resistance is graphed in Figure 3.2.

PNC316 steel has achieved irradiation performance which exceeds the level of Monju irradiation in material irradiation tests with the Joyo and FFTF reactors. Moreover, it had already been used in 27,000 cladding pipes for Joyo core fuel prior to its use in Monju, and its reliability had been demonstrated as a full production material.

3.3 Fuel Reliability Evaluations

In order to secure adequate fuel reliability in the face of operating condition fluctuations, operating fuel reliability tests were conducted in joint Japan-U.S. research using EBR-II. In tests conducted under over-output conditions, the fuel was found to have adequate safety tolerance relative to plant protection trip levels, and demonstrated not to rupture. Also, even under 60% - 100% of rated output operation in each operating cycle, no significant differences were found with the rated operation fuel. In tests in which ruptured fuel was continuously irradiated, the behavior of the reaction between the fuel and the sodium penetrating through the rupture was elucidated. It was evident from these tests that this reaction has a saturation point, and that the reaction products exist stably and inhibit outflow of fuel into the coolant.

As a result of these tests, it was verified that the Monju fuel was highly reliable. Also, with the detection of delayed-generation neutrons and FP gas, it was possible to achieve a high level of fuel rupture diagnostic technology.

4. Structure, Material

At Monju's operating temperature (529°C), the upper usable temperature limit (425°C in austenite-based stainless steel) set forth in the light water reactor structural design standard entitled "Bulletin Establishing Structural and Other Technical Standards for Power-Producing Atomic Energy Facilities" (MITI Bulletin No. 501) is exceeded, making it necessary to prepare a separate design standard for the high-temperature equipment. For this reason, R&D was done avidly on the high-temperature characteristics of the structural materials forming the foundation, on the environmental effects peculiar to high-speed sodium-cooled reactors, and on structural analysis and structural strength. This resulted in the formulation of the "High-Temperature Structural Design Guidelines for Type-1 Equipment in High-Speed Reactors" (hereinafter "High-Temperature Structural Design Guidelines"). Also, structural integrity evaluations were conducted on the supposition of the existence of an unlikely flaw, and earthquake integrity verification tests were performed on various pieces of equipment.

4.1 Structural Material Property Evaluation, Environmental Effects

Ascertaining the basic deformation behavior and mechanical strength of structural material, and the corresponding effects on the environment, is essential in designing high-speed reactor equipment. Monju's operating temperature would be roughly 30°C higher than Joyo's, making considerations of the creep phenomenon even more important in structural design. For this reason, the material testing facilities at Donen were beefed up, and a systematized material database was put together on domestically produced materials as pertaining to creep, stress mitigation, high-temperature fatigue, and creep fatigue. Based on this database, the various mechanical properties of the materials were formulated and the strength standards for materials used in Monju were established.

Experimental research was also done on the sodium environment effects and neutron irradiation effects on the structural materials, thus establishing evaluation procedures for use in the Monju design. In evaluating the effects of the sodium environment, what was of concern was the degree of corrosion loss due to sodium in the structural materials, and the deterioration of creep strength due to decarbonization through the sodium of the 2 ¼ Cr-1 Mo steel material used in the evaporator. In terms of neutron irradiation effect, the irradiation dosage limits at which tensile property integrity is maintained were determined, as were creep strength deterioration coefficients. Furthermore, in order to verify the changes in mechanical properties in the structural materials used in Monju in the face of neutron irradiation throughout the operating period, plans were made to take test pieces from critical locations in the structures, load these inside the reactor, and take them out periodically to subject them to tensile tests, creep tests, and other surveillance tests.

4.2 Development of Structural Analysis Techniques

High-speed reactor equipment is operated at high temperatures and subjected to pervasive thermal stresses, giving rise again to the need for structural analysis techniques for handling creep deformation and plastic deformation. For this reason, R&D was done on non-elastic analytical procedures, resulting in the perfection of analytical techniques which can be used in evaluating high-temperature strengths. Beginning in 1976, moreover, focusing on non-elastic analysis, Donen developed in-house a general-purpose nonlinear structural analysis program called FINAS as a means of conducting various types of structural analysis, including large-deformation and buckling analysis, dynamic response analysis, and heat-transfer analysis. This program was widely employed in establishing Monju evaluation and design-evaluation procedures.

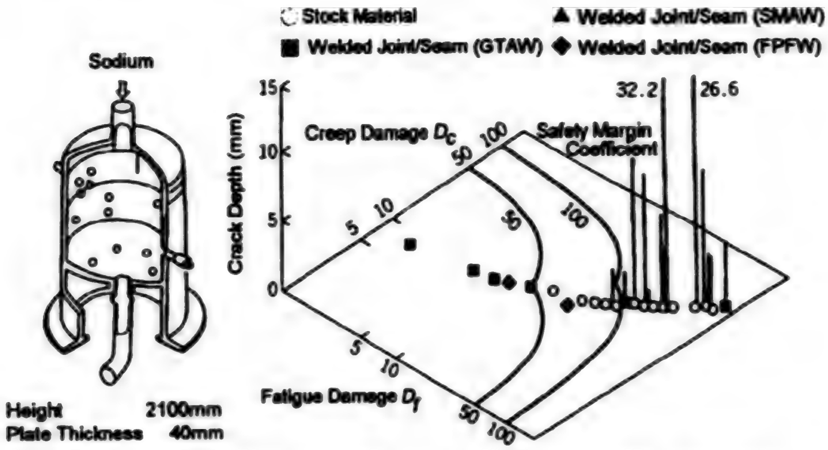


Figure 3.3 Thermal Transient Test Piece Creep Fatigue Evaluation

4.3 Structure Strength Tests

A wide range of fatigue tests, buckling tests, and heat ratchet tests were performed on pipe elements (straight pipe, elbows, T's, nozzles, etc.), assuming use in a high-speed reactor such as Monju. The findings were helpful in formulating pipeline design procedures.

Furthermore, in the structure strength verification testing facility (TTS), strength tests were conducted on vessels and other structures subjected to repeated thermal transient loads. As a result, the propriety of the "High-Temperature Structural Design Guidelines" discussed below, and the integrity of the structures fabricated with technology equivalent to that to be used in the actual Monju reactor, were established. In Figure 3.3 is depicted an example of the results of thermal transient strength tests done on structures simulating the characteristics of the reactor vessel and other primary structures. The objects tested were made with the same materials, fabrication methods, and welding procedures that would be used in the actual reactor structures. These tests demonstrated that the "High-Temperature Structural Design Guidelines" provide adequate safety margins for creep fatigue damage.

4.4 Formulation of 'High-Temperature Structural Design Guidelines'

When design work was first begun on Monju, the high-temperature structural design standard ASME Code Case, intended for application to high-speed reactors, had just been born in the United States, and the original idea was to follow this standard. However, as the Monju design was being evaluated, the U.S. standard was found to be inadequate as it stood, and it became clear that the evaluation techniques needed to be improved and made specific. For this reason, from 1977 to 1984, Donen worked with manufacturer specialists, integrating the latest R&D gains in structures and materials into their study, and came up with the "High-Temperature Structural Design Guidelines."

The "High-Temperature Structural Design Guidelines" did follow the basic ideas set forth in ASME Code Case N-47, but incorporated new and simpler evaluation procedures in view of Monju's characteristic operating conditions, and included supplemental provisions as well. Also, the standards for material strength that are included in the "Guidelines" are based on a database made up mainly of domestically made material data.

4.5 Structural Integrity and Virtual Flaws

In evaluating the structural safety of the Monju reactor, particular importance was given to the primary main cooling system pipelines. Assuming that an initial flaw occurred there, tests were done to investigate how the flaw would then develop under repeated loads. These tests showed that the developing behavior agreed well with the predicted values, that the length of the flaw would be small at the time of plate-thickness penetration, so that no unstable damage would result, and that the area of the actual rupture opening would be adequately small compared to the hypothetical design values.

4.6 Earthquake-Resistance Evaluation

Thoroughgoing attention has been given to the earthquake-survivability of equipment in Monju. From around 1972, vibration tests have been done on mockups of major pieces of equipment for the purpose of verifying the viability of the quake-resistance designs. These tests included reactor core structural element group vibration tests, reactor vessel vibration tests, sloshing tests, containment vessel vibration tests and buckling tests, primary cooling system pipeline vibration tests, tests to determine the insertability of the control rods during an earthquake, vibration tests on the out-of-reactor fuel storage tank, fuel replacement equipment, and steam generator, and function sustainability verification tests during earthquake conditions on pumps, valves, and other dynamic equipment. At the same time, suitable analysis codes were developed and employed in Monju's seismographic design.

5. Heat Flow

5.1 Heat Flow Inside Fuel Aggregates

Data necessary to the design of the fuel aggregates, such as mixing characteristics and pressure loss characteristics under conditions of high flow volume, were established in underwater and under-sodium tests on fuel aggregate models having differing fuel pin diameters, pin array pitches, and wire spacer winding pitches, etc. These were also reflected in the design code parameter selections, and used in optimizing the design of the geometrical shapes and pin arrays, etc., of the fuel aggregates, using analytical techniques.

When the coolant flow volume decreases transiently, producing a low flow volume like that of natural circulation, the coolant flow volume in the high-temperature regions inside the fuel aggregates increases due to buoyancy, so that a redistribution of flow volume inside the fuel aggregates takes place, resulting in the flattening of the temperature distribution in the radial direction. To analyze such transient heat flows inside the fuel aggregates, a single-phase flow sub-channel analysis code called ASFRE was developed.

5.2 Heat Flow Inside Reactor Vessel

An important consideration in designing the lower reactor core structure is that of distributing coolant flow volumes so as to keep the maximum cladding pipe temperatures about the same as that of the various fuel aggregates during rated power output. Techniques were developed based on a one-dimensional code using flow network models in order to select flow resistances in the flow-volume regulating mechanisms for this purpose. The propriety of the modeling methods and of the pressure loss coefficients used were verified in water flow tests on half-size models.

When an atomic reactor is scrammed and the output rapidly decreased, the temperature of the sodium flowing out of the reactor core drops precipitously. When this happens, low-temperature sodium flows into high-temperature sodium inside the upper plenum. As a result, a temperature-stratification phenomenon may develop in which high-temperature sodium stagnates at the top while low-temperature sodium flows over the bottom. In such a case, the walls of the reactor vessel and in-reactor structures are subjected to enormous thermal stresses due to the temperature gradient near the temperature layer interface. Thus it is of critical importance that the temperature-stratification phenomenon be subject to prediction and analysis.

For this purpose, the AQUA code was developed and perfected for analyzing multi-dimensional heat flows in single-phase, non-compressed fluids. This code is used in verifying the propriety of reactor vessel design and so forth. In Figure 3.4 is represented an example where the vibration at the strata interface observed in water tests, and the associated temperature fluctuations, can be precisely analyzed with the AQUA code using the high-order difference method. In experimental research, sodium tests are performed with 1/10 and 1/6 scale models, and water tests are conducted with 1/10, 1/7, and full scale 1/3-fractional models. The scale ratios between the actual article and the test model are selected so as to ascertain the overall phenomenon, combining tests so as to individually satisfy the laws of similitude between the phenomenon and the test model.

5.3 Heat Flow in Heat Transport System Equipment & Pipe

Research on heat flows in the equipment and pipe making up the heat transport system focused on tests to collect design data for the heat exchangers. Using the 50 MW steam generator test facility, 1/5-scale model tests were done on the intermediate heat exchanger, thereby verifying the propriety of the flow-volume distribution characteristics, heat transfer design formulas, and anti-thermal shock integrity. Steam generator characteristic tests were conducted with both 1 MW and 50 MW steam generators. In these tests, knowledge was gained concerning heat transfer flow evaluation formulas, flow

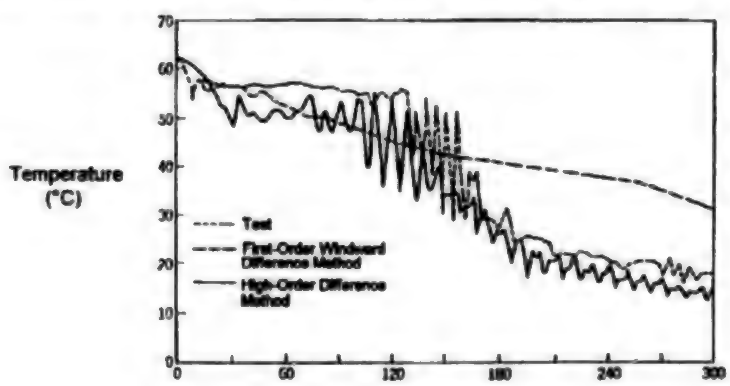


Figure 3.4 Temperature Fluctuations Associated With Vibrations at Temperature Strata Interface

volume distribution characteristics, transient thermal characteristics, blow-down characteristics, flow volume instability, and the dry-out phenomenon. For the air cooler(s) in the auxiliary cooling system, a full-scale model of the test article was placed in the 50 MW steam generator test facility, and characteristic tests were conducted. The results of natural circulation heat transfer characteristic tests were used in validating the design code called NATURAL.

5.4 Plant System Thermal Transient Characteristics

Using the 50 MW steam generator test facility in which one loop of the Monju was simulated to an output scale of 1/5, thermal transient characteristic tests—inclusive of control system action—were performed. The results were reflected in the design of the control system, and in studies on operating procedures, and also used in developing the dynamic characteristic analysis code COPD. Furthermore, in order to perfect a computation code with which the overall plant heat-flow dynamics could be analyzed for a wide range of conditions running from normal operation to mishap, the one-dimensional dynamic characteristic analysis code SSC was modified and used in analyses of coolant flow-volume reduction mishaps and large-bore pipe rupture accidents. System tests at Joyo and PLANDTL were employed in validating the analysis models.

6. Sodium Technology

R&D was conducted on purity control techniques, sodium analysis, techniques for moving radioactive materials in sodium and reducing exposure thereto, sodium combustion countermeasures, and tribological behavior in sodium, making use of the technology and experience gained in operating the Joyo reactor and with the sodium testing facilities at the Oarai Engineering Center.

6.1 Purity Control Technology

The cold trap that is used initially in the Joyo reactor to remove impurities from the sodium cooling system had an oxygen removal capacity of about 10 kg, which was inadequate for the Monju reactor. Thus a cold trap having an oxygen removal capacity of approximately 70 kg was developed by making the temperature more uniform inside the system, improving the design of the internal sodium flow channels, and optimizing the mesh filler ratio to sharply improve functionality.

Meanwhile, the analysis of such non-metallic impurities in the sodium as oxygen, hydrogen, carbon, and nitrogen, as well as of radioactive corrosive products (CP) and fissionable products (FP), is crucial in understanding substance transfer, corrosion behavior, and fuel rupture conditions. The things learned with Joyo were modified for Monju, and sodium sampling methods and various impurity analysis techniques were employed to make the analysis work more efficient.

6.2 Technology for Moving Radioactive Materials & Reducing Exposure Thereto

In order to reduce exposures during plant operation, maintenance, inspection, and repair, it is necessary to accurately ascertain the movement of radioactive species within the plant. The behavior of CPs and FPs in sodium systems is a phenomenon which involves both chemical and physical processes.

At Doden, radioactive species movement tests were conducted using an out-of-reactor loop. An analysis model of CP (^{60}Co, ^{54}Mn, etc.) behavior in sodium was developed, together with an analysis code called PSYCHE which was based thereupon. The propriety of this code was demonstrated by comparing the values calculated using this code with dosage rate distributions in the primary cooling system as measured in the Joyo reactor. SAFFIRE was developed as a code for analyzing FP behavior assuming a cladding pipe rupture, making it possible to predict FP behavior in the primary system. Reflected in this was knowledge gained in in-pile loop tests of FP behavior.

Meanwhile, efforts were being made to reduce the quantity of radioactive impurities in the sodium. These included suppressing the oxygen concentration in sodium (reducing corrosion), limiting the cobalt content of reactor core materials, and developing cobalt-free surface-hardened materials for use at control rod friction points and pump bearings.

6.3 Sodium Combustion Countermeasures

At Donen, sodium combustion behavior was studied in engineering-scale tests. These studies resulted in the development of combustion prevention measures and fire-extinguishing technology. In particular, while the structural reliability of the pipelines in the primary and secondary cooling systems had been thoroughly verified, rigorous measures were nevertheless developed for suppressing the combustion of sodium leaked onto the floor (onto the steel liner) in the unlikely event of a sodium leak from a pipeline. Furthermore, engineering tests were conducted on the combustion behavior of both spray-form and pooled sodium leaks, supposing a sodium leak from a pipe.

Based on these data, it was decided that the chambers in which sodium pipes and equipment would be placed inside the containment vessel would have a nitrogen atmosphere, and that the oxygen concentration therein would be continually monitored, so as to make it impossible for combustion to occur. Not only so, but sealed cellular structures would be used, with steel liners not only on the floor, but on the walls and ceilings of all concrete surfaces.

Most of the secondary system's pipelines and equipment are accommodated in the reactor's auxiliary building, which has an air atmosphere, but steel liners were to be installed on the floor, together with fire-suppression

tanks and so forth, so that even a sodium fire could be suppressed. Tests and analyses were done to verify the prevention of contact between leaked sodium and building concrete.

For the thermal insulation material used in Monju, an aluminum-oxide-based material was selected and is being used which, based on tests, excels in retarding sodium combustion.

6.4 Tribology in Sodium

Fuel aggregate wrapper-pipe pads (band-shaped cushions provided on the exterior surfaces of wrapper pipes to maintain the intervals between adjacent wrapper pipes) sometimes come into contact with each other due to fine vibrations caused by sodium flow or seismic activity. In the Joyo reactor, hardened chrome plate was used to prevent wear due to contact in sodium. In Monju, however, after conducting tests and research using simulations of the actual conditions, chrome-carbite explosive flame-coating was employed so as to stand up under protracted use under high temperatures.

It is possible that wear marks can develop due to friction vibration between the fuel cladding pipes and the wrapping wire. The influencing factors and growth behavior of such were studied in model tests in sodium, and it was found that limiting the gap between fuel pins within a suitable range is effective in suppressing this phenomenon. These findings are reflected in the Monju reactor design.

7. Equipment

The evaluation tests on the performance, functions, and durability of Monju's main equipment, building on the development experience gained with Joyo, involved the fabrication of partial or scale models and mockups. The main items are noted in Table 3.1. Mockup models were made of such moving equipment as the control rod drive mechanism, fuel exchangers, and main circulation pump. Partial or scale models were used for the other equipment. The equipment thus tested included the reactor structure, control rod drive mechanism, fuel handling system equipment, main circulation pump, and steam generator.

Table 3.1 R&D Model Dimensions for Main Equipment in Joyo and Monju

System/Equipment		Joyo	Monju	Remarks
Reactor Structural Systems	Reactor vessel	1/1	Reactor wall insulation	
	In-reactor structure	1/1	1/2	Water flow tests
	Upper reactor structure	1/1	1/6, 1/10	Flow tests
	Shielding plug	1/1	Diam: 1/2.5	
	Control rod mechanism	1/1	1/1	Tests in H_2O, Na
Cooling System	Pump	1/1	1/1	Tests in H_2O, Na (Impeller reduced in size in Na tests)
	Steam generator	Not in plant	1/5	Actual size dimensions 1/5 number of heat transfer pipes
Fuel Handling System	Fuel exchanger (internal)	1/1	1/1	
	Fuel canistering device	1/1	1/1	Excluding trolley
	Revolving plug	1/1	—	
	Extern. fuel storage tank	Not in plant	Part. model*	*Natural cooling only

7.1 Reactor Structure

In designing the reactor vessel, it is necessary to consider carefully the thermal stresses which, both during normal and transient conditions, are caused by axial temperature gradients at liquid surface levels on the reactor walls. The thermal conditions in Monju are different than in Joyo, so, in addition to employing thermal shielding plate to protect the reactor walls, a Y-shaped bucket structure was adopted in the vicinity of the liquid surface in the reactor vessel. Therefore, by employing the so-called two-liquid-level control method which controls the positions of two liquid surfaces in response to start-ups, shutdowns, and scrams, the temperatures in the radial direction were flattened. This approach is unique to Monju, so mockup models were used in tests under various operating conditions to ascertain the thermal characteristics and verify integrity.

For the reactor inlet/outlet nozzles and in-reactor structural supports, scale models were used in durability verification tests in the face of repeated thermal transients. For the shielding plug, scale models were used in thermal insulation performance tests. The temperature distributions and heat radiation amounts were evaluated, and the effectiveness of a mechanism to prevent natural convection of the cover gas was verified.

7.2 Control Rod Drive Mechanism

The control rods in Monju are divided between the main reactor shutdown system (including fine adjustment and coarse adjustment rods) and the backup reactor shutdown system (backup reactor shutdown rods). It is necessary that the linkage between the control rods and the drive mechanism therefor moves precisely. To this end, mockups were fabricated, and, under conditions simulating the actual hardware (i.e. in high-temperature sodium and in argon gas containing sodium vapor), various tests were performed, verifying that the functional and endurance requirements are satisfied. Furthermore, underwater quake-simulating vibration tests were also conducted, confirming that the functional requirements are met.

7.3 Fuel Handling Equipment

The offset-arm approach is employed in Monju's fuel exchangers. Mockups were designed and fabricated, and then tested in sodium to verify their action (gripper mechanism performance and aggregate insertability). Moreover, concerning the method of washing the sodium off after operation of the fuel exchangers, alcohol washing was used inside the containment vessel in the Joyo reactor, but this was found inadequate in the Monju reactor due to differences in the mechanism and the larger scale involved. What was decided on was the adoption of steam cleaning outside the containment vessel.

A method involving wrapping in stainless steel tape was adopted in the fuel handling system for the mechanisms that take the fuel aggregates that are contained in sodium pots outside of the reactor. In order to prevent mechanical failure due to sodium adhering to the tape when this method is employed, a device is attached to remove the sodium.

7.4 Main Circulation Pump

The main circulation pump in Monju is an upright, mechanical, centrifugal pump, which has a free liquid surface inside the casing, produces a flow volume of approximately 100 m^3/min, making it a much larger pump than that in Joyo (approx. 20 m^3/min). Mockup tests were conducted in water and in sodium to evaluate performance and durability. In the course of development, convection prevention plates were installed between the inner and outer casings to prevent thermal deformation of the shaft due to natural convection in the cover gas.

7.5 Steam Generator

The steam generator (SG) has a great impact on the safety, reliability, and economy of a high-speed reactor. Thus it was the subject of wide-ranging R&D. The Joyo reactor had no steam generator. Steam generator research was done from the beginning with Monju in mind.

The first tests, conducted with a 1 MW steam generator test facility, were aimed at evaluating the heat-transfer performance of the helical-coil heat transfer pipes. These tests demonstrated the viability of the helical-coil steam generator concept. Next, a 50 MW steam generator test facility was constructed, together with the successive fabrication of a No. 1 unit and a No. 2 unit, and trial operations were conducted. Tests were done in these trials on static characteristics, water flow safety, dynamic characteristics, overall cooling system controllability, heat transients, and mishap behavior. Further tests were done to simulate the operating procedures in Monju, to study hydrogen behavior by filling the sodium pipelines and steam generator trunk with hydrogen or water, and to demonstrate the viability of the water leak detection system. All these results were reflected in the Monju design. Also demonstrated were the pulling out of the pipe bundles, using the evaporator(s), cleaning, the heat transfer pipe plug(s), and the replacement of the outermost helical-coil heat transfer pipe. After all of these test operations, the steam generator was removed, disassembled, and examined, thereby verifying its integrity.

Related analysis codes were also studied and perfected in 1 MW and 50 MW steam generator tests. These included POPAI (for static characteristic analysis), COPD (for dynamic characteristic analysis), and BOST (for waterside flow safety analysis). These codes were used in designing Monju.

In the Monju steam generator, heat is transferred to the water and steam from the sodium via heat transfer pipe walls that are only 3 to 4 mm thick. The equipment must therefore be designed to handle a sodium-water reaction in the event of a heat transfer pipe rupture. Hydrogen meters are installed for early event detection, while a pressure release system and other safety equipment are installed to suppress the effects of such an event. Tests and analyses indicate that the reactor facilities exhibit adequate integrity even in the face of the largest water leak imaginable.

8. Safety

Comprehensive safety research has been performed. Preparing for the unlikely occurrence of an abnormal condition of mishap, the safety systems must identify and correct whatever events could have been the cause, and the course of events leading from that cause predicted and elucidated through tests and analyses. This was done, together with research to verify and demonstrate the effectiveness and performance both of preventative measures against abnormal events or mishaps and measures to mitigate the effects thereof.

More specifically, research was done on sodium leak events and sodium-water reaction events, etc., as design

standard events affecting safety, and research was conducted on such severe accidents as reactor core damage mishaps and related coolant (sodium) boiling mishaps. The research also included probability safety evaluations (PSA) in which the frequency of occurrence of events and the risks were analyzed. We here introduce the reader to some typical samples of this research.

8.1 Research on Sodium Leak Events

Concerning sodium leak events, research was conducted on leak conditions, the integrity of cell liners to prevent contact between sodium and concrete, and the behavior of sodium-concrete reactions in the event of a cell liner rupture. Research was also done to demonstrate the effectiveness of facilities installed to handle sodium leaks inside the secondary cooling system building.

To determine the effects of sodium leak events, research was also done to elucidate both the behavior of the aerosols produced and the effects of the aerosols on the equipment.

The sodium combustion analysis codes developed include SOFIRE (pool combustion), SPRAY (spray combustion), and ASSCOPS (mixed spray and pool combustion). In order to elucidate aerosol condensation and precipitation behavior, the ABC code was developed. In developing the analysis codes and the equipment to dampen the effects of a sodium leak event, use was made of test results gained with large test facilities such as SAPFIRE (Donen), ESMERALDA (CEA), and FAUNA (KfK).

8.2 Research on Sodium-Water Reactions

In the safety design of the steam generator, consideration was given to the possibility of leaks resulting from the complete and instantaneous failure of both ends of a heat transfer pipe. The SWACS code was developed to analyze the behavior of pressure developed from the sodium-water reaction that would result from such a failure. Tests were also conducted to obtain verification data. Then analysis and evaluation were performed, using these results. The integrity of the heat transfer pipe walls in the intermediate heat exchanger(s) constituting the interface between the primary and secondary system, and that of the coolant boundary in the secondary system were verified.

8.3 Research on Reactor Core Failure Events

In evaluating safety in the context of localized reactor core mishaps, in addition to pin contact and FP gas release abnormalities, also studied were events in which it was presumed that fuel aggregate flow was locally blocked by foreign matter. Related safety research included, in out-of-reactor tests, observations of reverse obstruction flow and evaluations of cooling impairment conditions, and, in in-reactor tests, the Mol 7C project and SCARABEE project conducted through joint international research. This research produced information on mishap expirability and on the FP release characteristics after a fuel pin failure. These studies confirmed that fuel aggregate cooling characteristics are preserved even in such cases.

Research was conducted under conditions of a hypothetical failure affecting the entire reactor core, with the mishap sequence represented in Figure 3.5.

In order to learn about fuel failures and post-failure substance behavior that strongly affect the course of events in the causative process, in-reactor tests at TREAT and CABRI were participated in, in the context of joint international research, and the causative-process analysis code SAS 3D was modified and perfected.

The sequences for cases in which the mishap does not terminate in the causative process can be divided into two categories. In the first category, the reactivity insertion in the causative process is not large enough to reach prompt criticality, and so first there is a transitional process in which the reactor core gradually melts down,

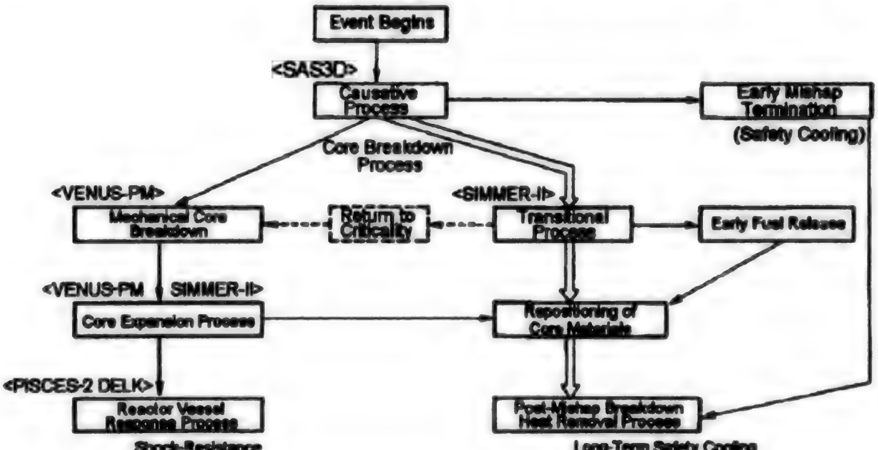


Figure 3.5 Hypothetical Reactor Core Breakdown Mishap Sequence

followed by a process in which the heat of decay is removed in a non-critical state. In the second category, there is reactivity insertion which exceeds prompt criticality in the causative process; first there is a mechanical reactor core breakdown process in which the core is rapidly heated and scattered, then the high-temperature reactor expands and generates mechanical energy in a core expansion process. In order to elucidate these event trains, various tests have been conducted, including out-of-reactor tests with JET and MELT, etc., on thermal damage, in-reactor tests using TRAN and JDBP in joint international research, and analyses using the whole-core damage analysis code SIMMER. These tests and analyses were used in studying the repositioning of scattered and solidified material from the melted-down core, and on the removal of decay heat from fuel debris.

As a result of these research efforts, using the SAS-SIMMER code, broad studies were done, ranging from the causative process to the reactor core breakdown process. These were rational studies, based on mechanics theory. By evaluating mechanical failure and damage using the out-of-reactor test VECTORS, as well as reactor vessel shock-resistance tests and the analytical code PISCES, it was made evident that the energy produced would be mitigated by sodium foam condensation and such physical phenomena as energy absorption by the in-reactor structure.

Research was also conducted to determine what would happen if a thermal or mechanical failure of the coolant boundary resulted in events that extended their effects outside the reactor vessel. The subject of this research was the behavior of FP aerosols released into the containment facilities. In research on mechanical and thermal transient responses of the containment facilities, the important subjects to study are sodium combustion, sodium-concrete reactions, and the interaction between molten substances and concrete. The CONTAIN code was developed for source term evaluation inside the containment facilities, taking these events into consideration. The application of this code proved it to be have high analytical functions.

8.4 Research on Coolant Boiling

Out-of-reactor tests were conducted on coolant (sodium) boiling using the transient boiling test facility SIENA, the breakdown heat boiling test facility DHB, and the plant transient response test facility PLANDTL, and databases related to two-phase flows and boiling heat-reduction limits were built.

In analysis research, a subchannel analysis code SABENA was developed, based on a two-fluid model, and this code was employed in evaluating out-of-reactor tests at Donen and KfK, as well as SLSF in-reactor tests. It was found possible to accurately simulate dry-out behavior as well as expansion and contraction behavior in the boiling region.

8.5 Probability Safety Analysis (PSA) Research

Applications research was conducted on probability safety evaluation techniques suitable to Monju, and development work was done to obtain the necessary database(s), system analysis techniques, and event trend analysis techniques. In the joint Japan-U.S. high-speed reactor equipment reliability database concentrated control organization CREDO, operational data from Joyo and FFTF, etc., are accumulated. Also, in developing system analysis techniques, various analytical programs were developed based on fault-tree analysis codes and event-tree analysis procedures. In event trend analysis, the functions of the previously listed safety analysis codes were expanded and employed. According to research cases so far, the risk of operating a high-speed reactor plant is satisfactorily low, and the fast reactor is found to be no riskier than the light water reactor.

9. Using Joyo

We next discuss natural circulation tests, FFDL tests, and fuel-material irradiation tests as specific examples of Joyo utilization.

9.1 Natural Circulation Tests

Natural circulation tests were conducted in stages from 0.5 MWt to 100 MWt for the purpose of verifying plant-specific safety (ability to remove decay heat by natural circulation) and to establish safety analysis procedures (natural circulation evaluation techniques) for the prototype reactor and future reactors.

There are two types of test, assuming constant reactor output. One is a normal test over the transition from forced circulation to natural circulation. The other is a transient test in which the decay heat is removed by natural circulation after a scram from a high-output state.

During natural circulation, there is not only an extremely low flow volume, but the in-reactor and in-system heat and movement exhibit complex behavior. Elucidating this behavior is critical to fully understanding natural circulation phenomena. In order to quantitatively analyze the behavior of natural circulation in the reactor and in the systems, tests and verification analyses were performed using dynamic plant characteristic codes (MIMIR-N2, COMMIX-1A, etc.), and analysis models were improved based on the test results.

Based on these tests, the capability of a fast reactor plant to remove decay heat by natural circulation was demonstrated. Furthermore, demonstration data were obtained for elucidating the important factors in determining the analysis procedures (radial heat transfer inside aggregates, etc.), and for validating the analysis codes (such as NATURAL) used for Monju.

9.2 Fuel Failure Detection, Location Determination (FFDL) Tests

Monju, as an FFDL, employs the tagging (or *tugging*) method. In developing an FFDL which uses the tagging method, one must develop the tag gas capsules and the tag gas enrichment and collecting unit. As part of these tests, Monju tagging simulation tests were done at the Joyo plant.

In these tests, use was made of an open device in which a shape-memory alloy developed for Monju was used in the tag gas capsules, together with an activated charcoal deep-cooling adsorption unit for enriching and collecting the tag gas. The functions of the tag gas capsules were verified, the constants needed for designing the activated charcoal deep-cooling adsorption method were determined, and the optimal operating conditions for the enrichment-collection unit were found. It was also demonstrated that the tag gas released into the cover gas could be enriched to a concentration that would allow the tag gas to be analyzed.

Meanwhile, concerning the development of the tag code, the main part of the code was found to be the simulation of tag gas composition variation induced by neutron irradiation. The code was modified and validated based on tag gas irradiation data obtained at the Joyo plant.

9.3 Fuel, Material Irradiation Tests

Irradiation tests were performed at Joyo on Monju's fuel, control rods, and structural materials. Tests were conducted on the fuel to verify over-output integrity, output rise pattern integrity, and high burn rate. Burn demonstration tests were done on the control rod materials, while surveillance backup tests were done on the structural materials. These tests provided irradiation data necessary to design evaluation and licensing.

In many other areas the technological gains made with Joyo are being reflected in Monju's design and technological development. These include technology for suppressing and removing radioactive corrosion products, technology for disposing of waste liquids including sodium, sodium vapor deposition suppression technology, and other knowledge gained in handling sodium for more than 10 years, as well as reactor core control codes, codes developed and validated at Joyo for operating and maintenance support systems, and operating and maintenance standards, system categories reflected in Monju, and other codes developed for Monju (shielding computation codes, dynamic plant characteristic codes, etc.).

Construction

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[FBIS Translated Text]

1. Site Work & Building Construction

1.1 Site Work

The site work got underway in January, 1983. This was divided into the following five operations.

(1) New construction and improvement (including prefectural highway improvement) from Sata to site.

(2) Ground preparation at temporary off-site grounds.

(3) Land construction including site preparation, building foundation excavation, underground channel for mountain streams on site, water access and discharge facilities

(4) Sea construction including building of ground retaining structures, breakwater levee, and loading pier

(5) Fabrication of various caissons and blocks at Technoport Fukui (old name: Fukui Rinko), 60 km from site

These land and sea construction operations would have to be implemented in terrain surrounded on three sides by mountains, making it necessary to conduct the work in a highly coordinated fashion. The volume of dirt excavated during the site preparation work would reach 2.3 million cubic meters, 1.0 million cubic meters of which would be used as fill, and the remainder of which was to be effective used in ground preparation at the temporary grounds. The sea work, due to sea and weather factors, would be limited to the 6 warmer months of the year. To move this work along expeditiously, a temporary dike was built, and a pocket was set aside in the land reclamation area to dispose of the earth and rock excavated during the winter months. The main site work operations are listed in Table 4.1

Table 4.1 Main Site Work Operations

Site Preparation	Site elevation	Sea level + 5.0 m to 42.8 m
	Site area	360,000 m^2 (facility site: approx. 90,000 m^2)
	Earth volumes	Excavation: 2,300,000 m^3 Banking: 1,200,000 m^3 Fill: 1,200,000 m^3
Revetments	Breakwater revetment (490 m)	Sloped levee 160 m Caisson-type hybrid levee 324 m
	South revetment	Block-type hybrid levee 140 m
	Harbor revetment	Wave-dissipating caisson upright levee 77 m
	Water access revetment	Caisson-type upright levee 127 m
Piers	Loading pier	Wave-dissipating caisson upright levee 163 m for 3000 DWT
Breakwaters	Breakwater	Caisson-type vertical levee 320 m
	Wave-retarding levee	Wave-dissipating caisson upright levee 30 m

1.2 Building Construction

The main buildings, in their location, size, and structure, are analogous to a 1.2 million kW class pressurized water reactor power plant. The main buildings to be constructed were the reactor building, the reactor auxiliary

building, the maintenance and waste disposal building, the diesel building, and the turbine building, the figures for which are summarized in Table 4.2. The main factors determining the structural design of these buildings were seismic activity, and the temperature of the coolant (sodium) in the unlikely event of a coolant leak.

Table 4.2 Main Building Summary

	Reactor Bldg. Reactor Aux. Building	Maintenance & Waste Disposal Bldg.	Diesel Building	Turbine Building
Building area (m^2)	11,400	3,000	1,400	3,300
Floor space (m^2)	50,800	14,000	5,500	14,000
Number of floors	2/4 undgrnd.	4/4	3/1	3/2
Outer dimensions (m)	100.0 x 115.0	47.8 x 57.5	36.5 x 38.5	38.1 x 34.6
Height (m)	34.8	30.2	20.2	17.0
Underground depth (m)	37.8	23.3	5.3	13.8
Structure				
Building proper	RC,ptl SRC,S	R,ptl SRC,S	RC	S, ptl RC
Foundation	RC	RC	RC	RC
RC: Reinforced concrete structure; SRC: Steel girder reinforced concrete structure; S: Steel structure				

It was verified that the buildings would withstand seismic shocks, both from the strongest earthquakes and from limited earthquakes, with building response falling within allowable ranges both in terms of flexibility and plastic-flexibility analysis. In addition, when the bodies of the reactor building and reactor auxiliary building were completed, vibration tests were conducted, and the propriety of the analysis results confirmed.

As to heat resistance, assuming the occurrence of an unlikely reactor coolant (at approximately 530°C during plant operation) leak, building integrity must be maintained even under such temperatures. Thermal stress analysis, crack analysis, and other analyses were done, together with various model tests, to demonstrate that this was the case.

Monju construction work got underway in October, 1985, with the foundation excavation work for the main buildings. In February, 1986, the mat concrete was poured for the reactor building and reactor auxiliary building. In August, 1986, the mat concrete was poured for the turbine building and diesel building. The construction progress for the main buildings is diagrammed in Table 4.3.

Some particulars pertaining to the enforcement of timeframe, quality, and safety considerations with the construction work are noted below.

(1) Use of large side cranes

(2) Efficiency enhanced with construction methods combining multi-purpose scaffolding and tower cranes

(3) Adoption of monitoring system to prevent interference between multiple crane groups

(4) Adoption of deck plate frame method and lining material frame work method

(5) Improved quality and shortened timeframe by use of steel girder platforms

(6) Efficient concrete pouring using distributors

(7) Improved quality and shortened timeframe by aggressive use of pipeline units

(8) Timeframe enforcement by aggressive use of preinstalled equipment

A year and a half after the construction work was started, construction of the reactor building, that is, the external shielding building and containment vessel, were completed. The reactor auxiliary building, maintenance and waste disposal building, turbine building, and diesel building were all completed in 1991.

2. Equipment Fabrication

2.1 Reactor Vessel Fabrication

The reactor vessel is made of SUS304 steel. The upper flange has a diameter of approximately 8.7 m. The overall height is 17.8 m, and the trunk plate thickness is approximately 50 mm. Forged ring material was used as the reactor vessel material to avoid longitudinal seams. As a result of employing the largest forgings which can be made domestically, the reactor vessel is configured of 12 forged ring units. As mentioned already, the reactor vessel's plate thickness is thin relative to its diameter, and high dimensional precision is demanded for reasons involving the control rod drive mechanism. Accordingly, in assembling the vessel, the upright placement assembly method was adopted in the interest of simplified assembly operations and a minimum of deformation during assembly.

In employing the upright placement assembly method, (1) an automatic horizontal side-facing welding technique was developed to enhance quality, minimize warp,

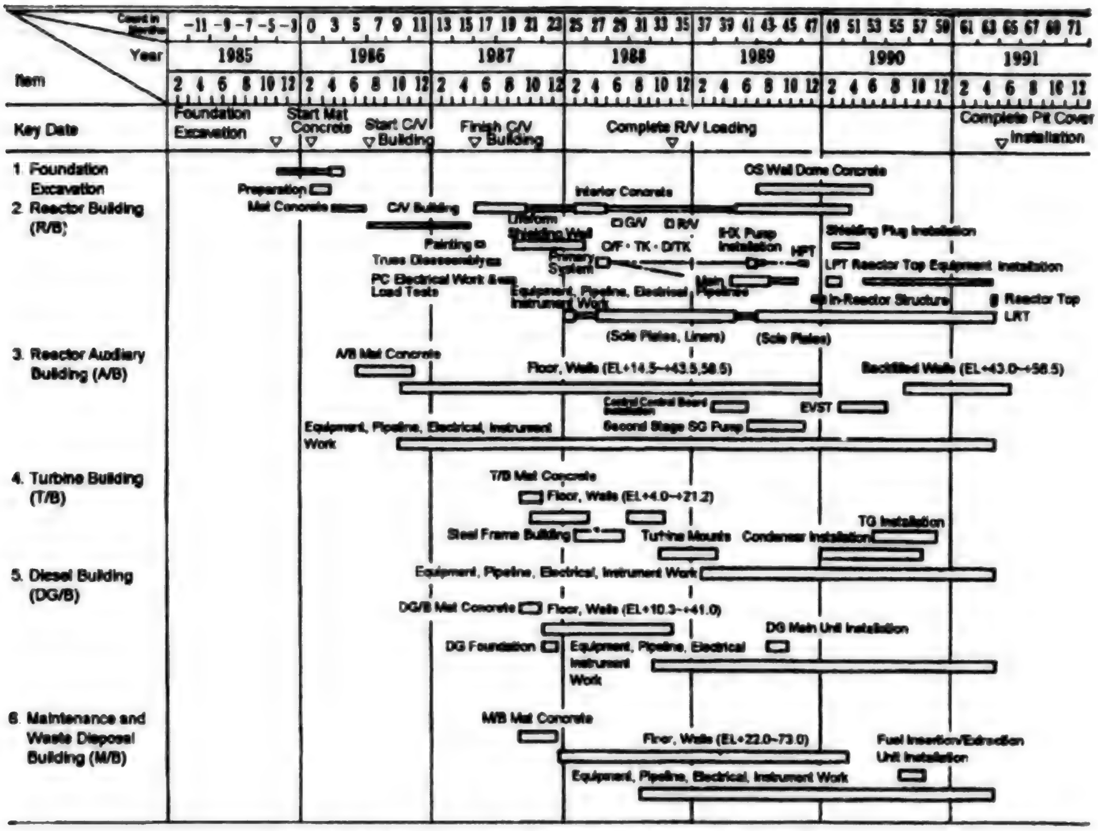


Table 4.3 Construction Progress for Main Buildings

and keep the seams running around the trunk horizontal, and (2) large machine tools were brought in which permit machining in the upright placement configuration. The side-facing Hot-TIG automatic welding method was developed and used. This is a hot-wire method in which the added wire is heated. It permits single-layer single-bead welding using a tip with a narrow opening. This method provides the same degree of efficiency as the SAW (submerged arc welding) method conventionally used in welding vessels. For stainless steel welding, it produces low entrance heat and low warp, and secures the integrity of high-temperature creep characteristics in the welds.

2.2 Shielding Plug Fabrication

The shielding plug is in effect the lid of the reactor vessel. It revolves to facilitate fuel replacement, and so demands high fabrication and operation precision. High assembly and installation precision are also required to insure good scram performance of the control rod drive mechanism. The shielding plug is a large piece of equipment, measuring approximately 9.5 m in diameter and having a total weight of approximately 950 tons. Nevertheless, it must move with high precision.

Every effort was made to reduce the seams in the extremely thick carbon-steel portions of the shielding plug, building the material up, while predicting weld warp amounts ahead of time using a mockup. Thus the welding and assembly were performed so that the finished plug would exhibit minimal warp.

The TILNAP method (magnetic oscillation narrow-aperture tip TIG automatic welding method) used in welding the thick stainless steel of the revolving plug shielding permits automatic welding, without moving the work, and offers the advantages of reduced welding steps and welding material using the narrow-aperture tip, and reduction of welding deformation. The thin-walled, large-diameter stainless steel members were welded in the upright placement condition, using special jigs to prevent deformation, and MC-TIL (magnetic oscillation automatic TIG welding).

In assembling the shielding plug, the revolving member is made up of a number of large components both in its stationary and moving parts. In order to assemble these components with high precision, prior to assembly, the components were put together and centered on a turning table measuring 16 meters in diameter, and held in position with knock pins. The knock-pin positioning approach makes it easy to secure reproducible assembly precision during on-site assembly and after post-start-up disassembly inspection. Since the equipment is so large,

the seals were checked for leaks as each piece of equipment was assembled, dimensions and shift measurements were made, and assembly was completed while conducting function tests.

2.3 Main Circulation Pump Fabrication

As the cooling system pumps, the main circulation pumps in Monju must exhibit satisfactory product quality. Great care and ingenuity were therefore employed in their fabrication, considering their importance to the plant systems. Specifically, these measures included:

(1) reduction of variance in fabrication precision between pumps (numbering three each in the primary and secondary systems), and achieving high centering precision in the revolving members by employing suitable special jigs, staging equipment, and joining processes,

(2) conducting vibration tests on the shafts under the same high-temperature conditions in which they would be used, taking meticulous care against bending, imposing strict bias-wall precision in parts exposed to air, and rendering imbalance correction easy by providing thick-wall spots at a number of locations in the radial direction,

(3) administering fabrication controls on dimensional tolerances stricter than the JIS standards for impellers, diffusers, and other components affecting performance,

(4) selecting the surface-hardened materials for bearings that exhibit reduced irradiation exposure during maintenance, establishing beforehand padding methods and flaw detection methods, and adequately securing the reliability of shaft sliding members, etc.

After fabrication, the actual control panels, rpm control units, main motors, and pony motors were assembled at the factory, and subjected to performance tests in room-temperature pure-water tests.

2.4 Intermediate Heat Exchanger Fabrication

In fabricating the intermediate heat exchanger,

(1) heat transfer pipe bundle fabrication and assembly techniques were adopted to achieve very high assembly precision in terms of strength and flow uniformity as compared to the heat exchangers used in light water reactors,

(2) high-quality welding techniques were adopted to facilitate operating with low entry heat and low warp, and to obtain good high-temperature creep characteristics in the welds.

In welding the interior shrouds and upper pipe plates, and the descending pipes on the secondary side and the lower pipe plates, welding techniques were employed in which an ingenious laminar procedure was used. Also, to make it possible to insure straightness, prevent deformation under its own weight, and ease of open tip alignment, the upright placement assembly method was used for the heat transfer pipe bundles. Heat transfer pipe insertion was performed in horizontal placement after the main members had been assembled.

The welding of the circumferential and longitudinal seams in the trunk and shroud was performed in the upright placement condition on a turning table using narrow-aperture tip Hot-TIG welding. Automatic TIG welding was used in welding the pipes and pipe plates after insertion of the heat transfer pipes. In order to even out the contraction deformation due to welding, the pipe plates were divided into several zones, and the welds were made alternately across the corners and in the interior zone. The welding of the secondary inlet plenum mirror plate after heat transfer pipe bundle assembly was performed in the horizontal placement condition using a rail-type automatic narrow-aperture tip welding unit.

2.5 Steam Generator Fabrication

The steam generator is made up of the evaporator and the superheater. We discuss here the fabrication of the evaporator. The evaporator was fabricated in three separate blocks, namely the upper plate section, the pipe bundle section, and the trunk section. Then these were assembled into an integrated unit.

The heat transfer pipe bundles are made of 2 ¼ Cr-1 Mo low-alloy steel, arranged in the form of a helical coil. The heat transfer pipes must be highly reliable as they form the separating boundary between the sodium and the water/steam. Thus their fabrication requires high-precision bending and assembling techniques.

The heat transfer pipe helical coil unit is first automatically welded from top to bottom as a straight pipe is turned, thus producing a long straight of the length required to fabricate the helical coil. This long straight pipe is then bent into a helical coil by a roller bender used in bending operations, set to the desired diameter, coil center, and winding pitch. While this is going on, the pipe is continuously built into an H-beam heat transfer pipe support structure. The H-beam steel becomes the structural center of the pipe bundle unit, together with the inside tube to which this is attached. The heat transfer pipe, in its helical coil shape, is inserted in holes in the H-beam steel and thereby supported. The H-beam steel and the inside tube must be fabricated with high precision as their dimensional precision affects the dimensional precision of the entire pipe bundle unit. Stress-removing annealing is performed on the H-beam steel material, and cutting techniques are used which impart little machining warp. Standard-surface machining is performed using special-dimensional formed steel. The coils are built up to 14 layers, sequentially from the inside. After they are supported and secured by the H-beam steel, the helical coil pipe bundle unit is stood up and assembled together with the upper plate.

The upper plate structure was adopted to facilitate the assembling of the rising pipe and descending pipe. The upper plate and the pipe bundle unit are held together with turnbuckles and inserted into the trunk. When this is done, however, the techniques used and the dimensional tolerances involved in centering the upper plate, pipe bundle unit, and trunk, and in positioning the radial keys which connect the pipe bundle unit to the trunk, are very important. After the positioning, in which each position is accurately measured, and the assembly, the ascending and descending pipes are sequentially put together and welded. When the integrated upper plate and pipe bundle units are inserted into the trunk, it is impossible, for structural reasons, to do this in the upright configuration, requiring a height of more than 30 meters. Therefore, at the factory, the units are suspended in an assembly pit in the floor for assembly.

The positions of the welding seams are determined on the basis of the heat transfer pipe shape and with regard to interference with other internal structures. However, at places where welding is done while sequentially putting members such as the ascending and descending pipe together, or where pipe groups are concentrated, as at the seams of pipe and pipe plate, the welding must be done in narrowly confined spaces. Thus butt joints are used for the pipe-pipe plate seams, with the operation proceeding from the inner pipe surface where placement of electrodes is relatively easy. After temporarily attaching the heat transfer pipes to pipe brackets bent out from the pipe plate(s), the welding torch is set at the inner pipe surface and welding is done by the TIG method.

2.6 Out-of-Reactor Fuel Storage Facility Fabrication

The fabrication of the shielding plug and revolving rack for the out-of-reactor fuel storage facility is now discussed. The shielding plug main unit is a large stainless-steel structure having a diameter of approximately 7 m and a height of approximately 1.3 m. Nevertheless, since it is employed in fuel handling, it requires high dimensional precision. For this reason, it is necessary to minimize welding deformation. Therefore, by fabricating a mockup, the amount of deformation was determined for each seal or joint shape, and every welding process, a half-scale model was used to pre-confirm the amount of deformation produced by each fabrication operation in order, and then the actual equipment was fabricated. Automatic MAG welders were used which have a multi-layer buildup feature.

The revolving rack is a large welded structure made of stainless steel, approximately 5.5 meters in diameter and 4.5 meters in height. As in the reactor vessel, the walls are thin compared to the diameter, so the fabrication procedures were determined so as to prevent welding deformation, thus insuring the requisite precision.

3. Equipment Transport

The large pieces of equipment and heavy materials installed in Monju were transported to the site by sea. The building site faces the Sea of Japan, so the months of April through October, offering mild weather and sea conditions, are preferred for sea transport, making careful coordination with the progress of construction necessary. The unloading was done using the loading pier facilities installed on the front shore. These facilities were completed in August, 1988. Prior to this unloading was done at a temporary pier.

In August, 1988, with the completion of the loading pier facilities, a 150 ton stiffleg derrick crane and a 550 ton gin pole derrick crane were installed, enormously improving pier operations.

3.1 Transport of Reactor Containment Vessel Members

The trunk plates, personal air locks, equipment conveyor entrance, polar crane, and other equipment used in the reactor containment vessel were shipped by sea from factories in Kobe and Hiroshima via the Kanmon Strait, making more than 20 shipments in all to carry approximately 5400 tons.

In shipping the secondary main cooling system containment vessel penetration module, rust-preventing nitrogen gas was sealed into the penetration module, and the exterior was covered first with a protective wooden cover, then with a waterproof tarpaulin (coated with tar waterproofing). The bellows unit configuring the penetration module was further protected with a covering of zinc plate. Thus rigorously protected, the module was delivered to the site (February, 1988).

3.2 Reactor Vessel and Guard Vessel

The reactor vessel guard vessel (approximately 470 tons) was housed in a shipment cask at the factory. This cask was then welded to the deck of a barge and the barge pulled by tugboat to the site. At the site, the guard vessel, still in its cask, was unloaded by a floating crane, and then rolled to the containment vessel (June, 1988).

The reactor vessel was packed in double layers of poly sheet and halo sheet at the factory, loaded horizontally onto a ship, and transported to the Monju harbor. At the site, it was lifted, in the horizontal position, by the gin pole derrick crane at the finished loading pier and offloaded onto rolling materials positioned above a drainage stage (October, 1988).

3.3 Cooling System Equipment Transport

The intermediate heat exchanger, evaporator, superheater, and air cooling equipment that makes up the auxiliary cooling equipment were all shipped by sea and delivered to the site between June and October, 1989. The evaporator, measuring approximately 15.5 m in length and 4.5 m in width, weighed some 170 tons packed, so the stiffleg derrick crane could not be used, and it had to be offloaded with the gin pole derrick crane. The evaporator was covered with a three-ply cover sheet during sea transport and offloading to prevent salt contamination. The cover sheet was then peeled off gradually during installation. To prevent rust, nitrogen gas was

sealed inside the superheater at the factory prior to shipment. The pressure of this nitrogen gas was monitored during transport by sea, and gas cylinders were carried onboard the ship in the event of a pressure drop.

The main circulation pumps, both for the primary and secondary systems, were shipped by land, separated into inner and outer assemblies.

4. Equipment Installation

Serious construction work on Monju got underway in October, 1985, beginning with the foundation excavation. In February, 1986, the mat concrete pouring began for both the reactor building and the reactor auxiliary building, and work began on the reactor containment vessel in July. Steel plate blocks enlarged in a temporary on-site factory were assembled one after the other, and had been completely attached to the containment vessel in April, 1987. In June, 1988, the reactor vessel guard vessel was installed, followed by the main reactor vessel unit in October of that year.

Another critical operation was the laying of some 4 million meters of electrical cable. This work, done amidst intense construction operations, was finished within the original timeframe.

On 25 April 1991, the equipment installation work, as based on the initial planning, was finished with the installation of the last pit cover positioned on the operating floor above the reactor vessel. Then, following on 18 May, a complete inspection was done in preparation for the beginning of comprehensive function tests.

The equipment installation work was based on a master schedule. Then, from the master, more detailed site master schedules were sequentially drafted for roughly the next one year. The work was done according to these master schedules in the interest of insuring high-quality construction and safety. More than 20 subcontractors were employed, making it necessary to hold frequent coordination meetings.

With sodium used as the coolant in the Monju reactor, the sodium equipment and pipelines could not be washed with water after installation, making it necessary to be particularly careful in monitoring the cleanness of sodium-contacting surfaces in the sodium equipment. To this end, the "Sodium Equipment Cleanness Control Guidelines" were drafted, prior to construction. These guidelines, which reflected experience with Joyo, designated cleanness control zones, depending on the importance of the equipment and the construction situation, and aimed to approximate a factory work environment for on-site welding operations, etc. Subsequently, these guidelines were extended to cover non-sodium equipment, reflecting light water reactor experience, and also the "Salt Damage Countermeasure Guidelines," so that they were used as the cleanness control guidelines for Monju equipment in general.

The construction work schedule for Monju is diagrammed in Table 4.4.

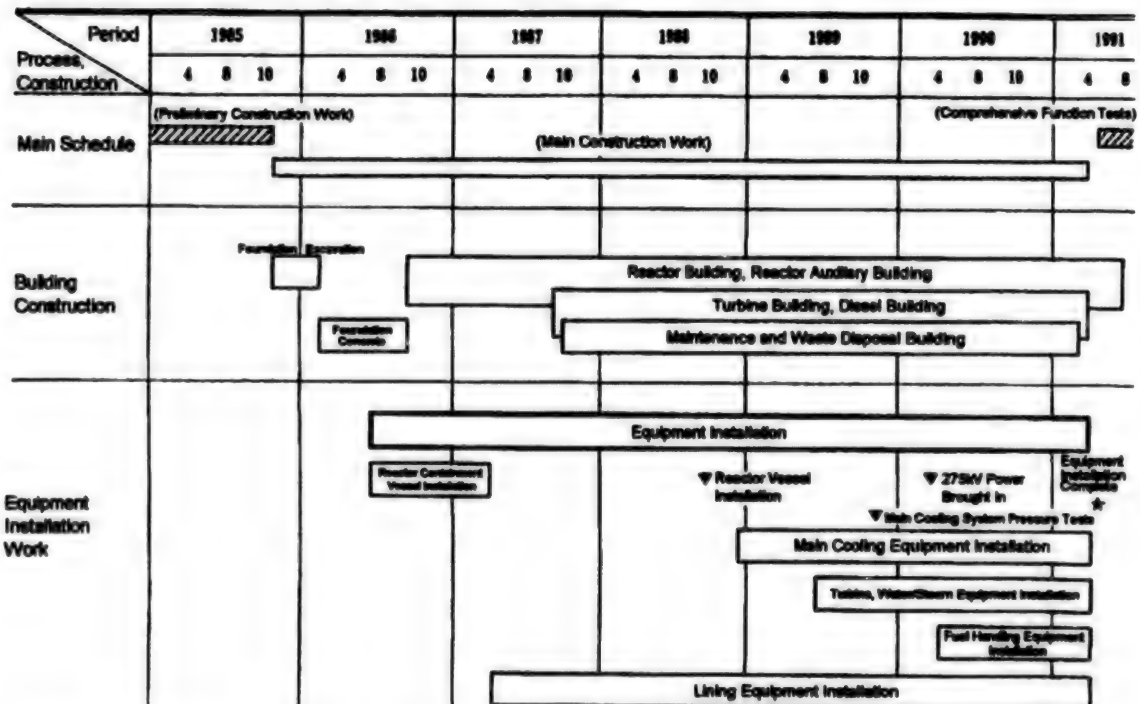


Table 4.4 Monju Construction Work Schedule

4.1 Reactor Containment Vessel Installation

The containment vessel is basically the same as in a light water reactor, except that it is the largest in Japan, measuring approximately 49.5 m in diameter and 79 m in height. The cylindrical steel containment vessel, which is hemispherical at the top and dish-form at the bottom, has six reinforcing rings around its girth. For installation, the top of the cylinder was divided axially into 13 segments. The installation work began in July, 1986, with the installation of temporary supports to temporarily support the weight of the entire vessel, and was completed in approximately 10 months.

4.2 Lining Equipment Installation

Lining equipment is installed in all rooms which house sodium equipment and sodium pipeline. The structure thereof falls into three categories. In room in which radioactive sodium is handled, as with the primary cooling system and fuel handling system, the entire room is given a sealed structure, and filled with a nitrogen atmosphere, so the lining equipment used is of the stationary and semifloating types, while an open catch-pan type is used for the secondary cooling system which contains non-radioactive sodium.

The lining equipment construction covered approximately 4 years at the site, running concurrently with the building construction, and actually, during this period, constituting one quarter of the critical operations in the overall construction work. In beginning this work, since it was recognized as critical from the beginning, efforts were made from the construction schedule study stage to streamline the work procedures. In the primary system lining equipment work, a process center covering a floor space of approximately 990 m^2 was provided on the site, and the steel plate liner panels were prefabricated on site to the extent possible, with great efforts made to insure safe installation, establish the fabrication process, respond quickly to site process needs, and improve both quality and work efficiency while shortening construction time. Many unprecedented innovations were implemented, including using the steel plate liners as forms when pouring the building concrete and streamlining the building processes. The installation of lining equipment in the secondary system must wait until after the building construction, so the equipment was made into blocks at the factory, thus improving workability during on-site construction.

4.3 Lifeform Shielding Wall (Reactor Vessel Chamber) Installation

The lifeform shielding walls are large steel plate and concrete structures having a hexagonal cross-section, an overall height of approximately 25 m, and a wall thickness of approximately 2.5 m. They are divided into four segments vertically. Each of these blocks was manufactured in multiple block form at the factory, with consideration given to the installation procedure. Each block in the lifeform shielding walls must support the reactor structure which has a total weight of approximately 10,000 tons, so it was given a welded structure using multiple reinforcing steel plates.

The lifeform shielding wall installation work was begun in May, 1987, and took some 16 months to complete. The blocks were shipped in by sea, then given their final dimensions at the site. Installation was performed sequentially on the foundation base inside the containment vessel which had already been installed. In installing these blocks, which were welded together during factory fabrication, care was taken to prevent welding deformation, with much automatic welding being used. In this way quality was secured.

4.4 Reactor Vessel Structure Equipment Installation

The reactor structure equipment is made up of the main reactor vessel, the reactor vessel guard vessel built exterior thereto, the in-reactor structure inside the reactor vessel, the shielding plug which corresponds to the cover at the top of the reactor vessel, and the large precision equipment that carries the reactor core top mechanism. High product quality and installation precision are demanded.

In installing the reactor vessel, it took about a week to roll the reactor vessel from the pier on the site to the interior of the containment vessel. In installing all of the other equipment, the Joyo experience was drawn upon, limitation value control procedures were enforced to raise both fabrication and installation precision, and thus high precision was achieved.

4.5 Large-Bore Sodium Pipeline Installation

In the primary main cooling system, the pipeline at the reactor outlet, that is, pipe approximately 80 cm in diameter with a wall thickness of 11 mm, has sodium flowing through it at operating conditions of approximately 530°C and 1.4 kg/cm^2G. Furthermore, in the secondary main cooling system, coolant that has exchanged heat with the primary system coolant flows through pipe that is 56 cm in diameter with a wall thickness of 9.5 mm, at operating conditions of approximately 505°C and 5.6 kg/cm^2G, on its way to the steam generator. The pipe used in making up these systems is thin-wall, large-bore pipe.

The laying of the pipelines in the primary main cooling system began in June, 1989, and took approximately 5 months, while that in the secondary main cooling system began in August, 1989, and took approximately 10 months. In installing the pipelines, restrictions were very strict in the work space. Automatic welding was used for all of the welding in the primary system, the pipeline of which is shorter than that in the secondary system, thus enhancing the quality and stability of the welds. Also, in the secondary system, the long pipelines were divided into blocks in the factory, thus reducing the amount of on-site welding work and further enhancing quality.

5. Receiving the Sodium

A total of approximately 1700 tons of sodium are needed for the Monju plant. This compares to approximately 200 tons used in the Joyo reactor. Following day orders and the trial operation schedule, this enormous quantity of sodium had to be shipped in, stored, and then charged, all while preserving its quality and insuring safety. The production of the sodium was ordered to the French company Metospecio. From the factory near Albertville, the material was shipped out of Le Havre harbor, through the Suez Canal, to Kobe harbor, and on to Tsuruga. The sodium was shipped in 92 18.5-ton containers, for a total of 1703 tons. The shipments were made without incident from 22 March to 7 November, 1991.

To store the sodium, two SUS steel tanks were temporarily installed in space on the west side of the reactor auxiliary building. Each tank—the largest made in Japan—held 300 tons. They were used as a plant supply buffer and for sodium quality checks. The quality of the sodium would deteriorate as long as there was a possibility of contamination, so composition controls were enforced, the tanks were thoroughly cleaned, and measures taken to prevent air getting in them. The sodium purity turned out to be less than 5 ppm of oxygen, which is good, at the time of factory shipment, intake into the storage tanks, and intake into the plant.

Between 1 July and 28 November, 1991, sodium was transferred to the plant systems in coordination with the testing schedule.

6. Fabrication of Initial Fuel Complement

The initial complement of fuel for the Monju reactor core (hereinafter "Monju fuel") was fabricated on the 3rd Plutonium Fuel Development Lab FBR line at the Tokai facility.

The development of plutonium fuel fabrication technology began on an experimental scale with various basic tests on uranium-plutonium mixed oxide fuel (hereinafter "MOX fuel") at the 1st Development Lab. Beginning in 1969, MOX fuel was fabricated for irradiation use overseas, including Saxton, Rhapsody, Donley, and Halden. Thus was technical know-how built up.

In 1972, the 2nd Development Lab was opened as a pilot plant type of facility for manufacturing MOX fuel. Since then, this facility has been manufacturing various types of irradiation-use fuel for the heavy water critical experimental unit (DCA), the new conversion prototype reactor Fugen, the experimental fast reactor Joyo, and other domestic and foreign facilities.

The 3rd Development Lab was built in 1987. Reflecting the knowledge gained in fuel fabrication for 20 years in the 1st and 2nd labs, this lab was built with future high-order plutonium in view. By using remote automation and dry processes, the facility aimed to reduce exposure, implement mass production, and reduce costs. Thus it was positioned as a facility for developing technology to practically implement MOX fuel fabrication. After construction, this facility was subjected to uranium test operations and then plutonium test operations. Beginning in October, 1988, the facility began to fabricate fuel for the fifth replacement in Joyo. After this it began to fabricate Monju fuel.

Monju fuel fabrication began in the fall of 1989. At the end of a long campaign which lasted approximately 4 ½ years, a total of 200 fuel aggregates (including spare aggregates) were manufactured, for use in the inner and outer reactor core and as test fuel aggregates.

Whereas the fuel pellets used in the Joyo, Fugen, and ordinary light water reactors were high density, having a theoretical density of 93% to 95%, the fuel pellets used in Monju were given a low density of 85% to reduce swelling during burning.

In manufacturing the low-density fuel pellets, an organic pulverizer called a density suppressant was mixed into the raw material MOX powder and this was molded in a press. Then, during sintering, the density suppressant was sublimated off by the sintering heat, so that innumerable air pockets were left in the pellets. The technology for fabricating pellets with this method began with the selection of the density suppressant. Repeated tests were done to optimize such controlling factors as admixture ratio, mixing and granulating method, and molding pressure. The data thus gleaned were used in perfecting the technology.

Also needed are techniques for handling large quantities of the small, lightweight pellets at high speed. To this end, a new handling mechanism was developed, and further modifications and improvements resulted in a high degree of skill. As a result of these efforts, it became possible to stably mass-produce low-density pellets.

In the Monju fuel aggregates, a mixture of stable xenon and krypton isotope gases (tag gas) is sealed into the fuel pins, with the composition differing for each aggregate. Thus, in the unlikely event of a fuel failure during irradiation, it is possible to identify the failed fuel element. For this reason, in assembling the fuel pins, when the pellets are packed in, tag gas capsules in which fuel failure detection gas is sealed are also loaded in. Then, after both ends of the pin are sealed shut by welding, the capsules are unsealed. This is an unprecedented innovation.

In order to pack the pellets in smoothly and enhance the end plug welding performance, other innovative improvements were made and reflected in the automated equipment, thus facilitating mass production.

Meanwhile, after the plutonium supplied for use in the Monju fuel is used, it is reprocessed at the Tokai Reprocessing Plant, then converted into MOX powder at the Conversion Technology Development Facility. In determining the degree of fuel plutonium enrichment, first the

composition changes due to Pu-241 decay during reactor criticality and fuel fabrication lead time were weighed, and then the "equivalent fissile method" was adopted so as to insure the requisite reactor core reaction rate. For this reason, in order to suppress variation in plutonium isotopes between raw material supply batches, the post-treatment plutonium nitrate solution was division-controlled, and various improvements were made to enhance the isotope analysis precision and raw material weight-mixture precision. Also, techniques have been developed at the Conversion Technology Development Facility for improving the denitrification efficiency.

In addition to the fuel fabrication and processing technology mentioned above, many technical developments were made in the areas of inspection, analysis, production control, security measures, and criticality management. In terms of security measures, in particular, the NDA (non-destructive assay) system was developed and widely implemented. This has yielded good results in improving the efficiency of security measures and has been commended by the International Nuclear Material Monitoring Society.

Furthermore, in the area of radioactivity control, in view of the facility's large size and degree of automation, remote-control and labor-reduction measures were taken with the introduction of remote concentrated control systems and self-propelled radiation monitors in each operation.

Moreover, throughout the entire fuel fabrication operation, ordered-product quality control guarantee provisions are established, and planning and control, work control, operation progress control, and quality control are rigorously enforced, thereby protecting against the shipment of defective product. As a result, it is possible to guarantee the quality demanded in each fuel pin and in each fuel pellet.

Thus, throughout the fabrication of Monju fuel at the 3rd Plutonium Fuel Development Lab, low-density pellet production technology is established, and remote automation technology is demonstrated. This obliges the conclusion that Japan can be proud at home and abroad of its MOX fuel fabrication technology, together with its technology for manufacturing high-density pellets developed earlier.

As to the future, it is hoped that the abundance of successes and knowledge gained thus far will be reflected in the manufacture of replacement fuel for the Monju reactor, as we look to the construction of a MOX fuel production plant sometime in the future, as well as further advances in technological development.

Test Operation

94FE0753E Tokyo GENSHIRYOKU KOGYO in Japanese Jun 94 pp 51-62

[FBIS Translated Text]

1. Overview

Monju is being subjected to trial operations to complete it as a highly safe and reliable prototype plant, and to demonstrate the technology that would lead to future reactors such as the fast demonstration reactor.

The objectives of the trial operations are to verify the functions of the systems configuring the plant, to demonstrate its safety and reliability, to check its design and evaluate its flexibility as based on the trial operations data, to provide hard data for developing fast reactors, and for training the operating personnel.

The trial operations can be divided between the comprehensive function tests to verify the functions of plant equipment prior to loading the core fuel, but particularly to verify the functions of systems connected with sodium and the total functions of the overall plant, on the one hand, and the performance tests to verify plant performance from core fuel charging to the normal operation stage, on the other. In Figure 5.1 are indicated the scope of Monju trial operations and the testing sequence.

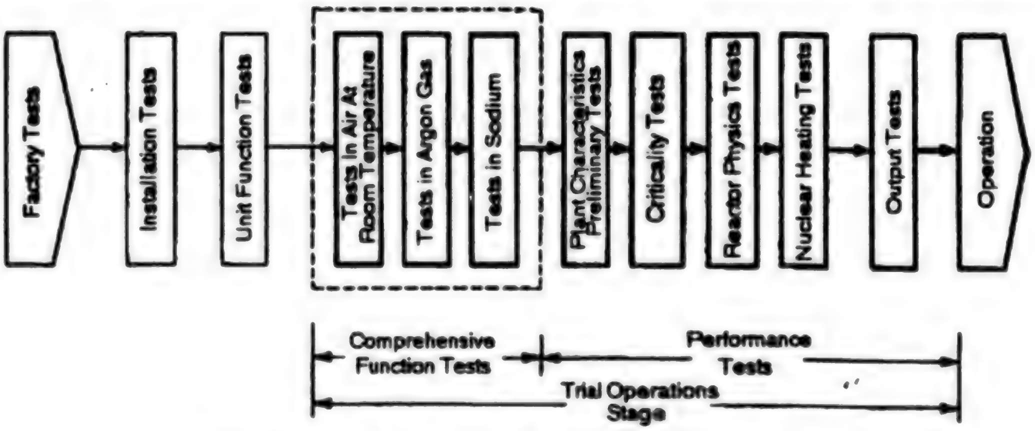


Figure 5.1 Monju Trial Operations Scope and Testing Sequence

Because sodium is used as the coolant in the Monju reactor, the comprehensive function tests were implemented in stages on systems connected with sodium, such as the first and second cooling system equipment, and the fuel handling and storage equipment, namely tests in air at room temperature, tests in argon gas, and tests in sodium.

The performance was verified in stages, from plant characteristics preliminary tests, to criticality tests, to reactor physics tests, to nuclear heating tests, and finally to output tests.

In the plant characteristics preliminary tests, plant temperature-rise conditions for the output rise tests are produced by introducing heat with the first and second main circulation pumps. In other words, by raising the temperature to the operating temperature on the cold leg pipeline side in the primary and secondary main cooling systems with the pumped heat, the plant's operating characteristics are preliminarily evaluated.

The criticality tests begin with the charging of neutron sources into the simulated reactor core configured for the comprehensive function tests. The dummy fuel aggregates are sequentially replaced with reactor core fuel aggregates until initial criticality is reached. After that, the initial core configuration is complete, and such things as excess reactivity are verified.

In the reactor physics tests, the control rods and fuel are subjected to reactivity value characteristic evaluations, reactor shutdown flexibility is verified, and the shielding around the reactor is evaluated. These measurements are made at very low reactor output. Radioactivated foil is used in collecting output distribution characteristic evaluation data.

In the nuclear heating tests, the system temperatures are raised by nuclear heating, the water/steam/turbine power generating equipment is subjected to trial runs and adjustments are made, and the initial entry is made into the electric power system.

After that, in conducting the output tests, the output level is raised in stages. At each output level, performance verifications are made of such things as plat system operating characteristics, control characteristics, and transient characteristics, and adjustments are made. As the output level is raised, measurements are also made on the radioactivity shielding and chemical analyses are conducted to verify the requisite performance.

After the performance verification tests have been completed at each stage, the reactor is run continuously at 100% output to test the loads, and to verify that the plant can be operated stably without malfunction.

2. Comprehensive Function Test—Planning

The purposes of the comprehensive function tests are to verify the functions and performance of the equipment and systems, to examine the propriety of the design specifications and standards, as well as analysis codes, as based on R&D findings, and to establish in which the core fuel can be charged.

The individual tests in the comprehensive function tests are large[illegible] divided between the various equipment groups, including reactor structure, reactor containment vessel, primary and secondary cooling system equipment, water-steam-turbine power generation equipment, fuel handling and storage equipment, radioactive waste disposal equipment, and measuring and control equipment. The number of these tests is approximately 300. The main test categories are listed in Table 5.1.

Table 5.1 Main Test Categories in Comprehensive Function Tests

Large Category (Equipment Category)	Main Test Categories
Reactor structure	Control rod drive mechanism function tests, reactor vessel ISI unit action tests
Reactor containment vessel	Reactor containment vessel leak rate tests
Primary, secondary cooling system equipment	Preheat test, Na fill & drain tests, Na auxiliary function tests, pipeline heat displacement measurement tests, pipeline vibration tests, main circulation pump characteristics tests, cooling system tests, ISI unit action tests, auxiliary cooling equipment function tests
Water/steam/turbine power generation equipment	Tests on water/steam system functions surrounding the steam generator
Fuel handling & storage equipment	Fuel replacement tests, fuel transport tests, fuel washing equipment function tests
Radioactive waste disposal equipment	Gas/liquid/solid waste disposal equipment function tests
Measurement and control equipment	Reactor security system tests, plant control system equipment function tests

Monju is Japan's first fast breeder power reactor. Thus, besides the tests that have been performed conventionally at nuclear power plants, there are other tests, such as the cooling system natural circulation tests, which are peculiar to Monju and the development of practical fast reactors.

Planning for the selection of the test categories included pre-utilization inspections pertaining to performance prior to fuel charging as based on regulations, verification of design performance and operating performance, collection of hard data for R&D, and providing operating experience to operating personnel.

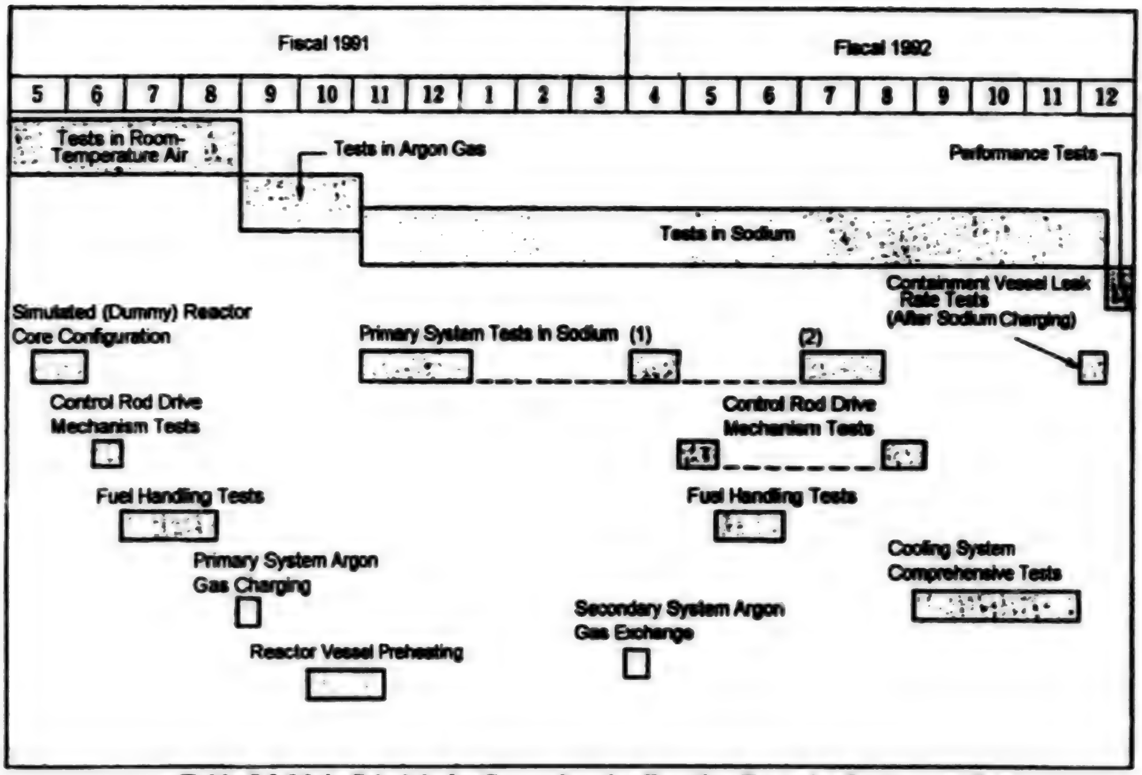


Table 5.2 Main Schedule for Comprehensive Function Tests (as Implemented)

After the installation of the equipment, for about 20 months, from May, 1991, to December, 1992, comprehensive function tests were conducted. The main schedule of the comprehensive function tests (as implemented) is given in Table 5.2.

3. Comprehensive Function Tests—Results

With the focus on the sodium cooling system equipment characteristic of FBR systems, we now summarize the results of the comprehensive function tests on the Monju plant.

3.1 Argon Gas Exchange

Before loading sodium into the sodium cooling system equipment, the air inside the systems are replaced with inert argon gas. This is done by repeated evacuations and clean argon gas charging. The ability to maintain the oxygen concentration inside the systems at adequately low levels is then verified, as is the high degree of sealing throughout all the systems.

3.2 Preheating

In the preheating of the reactor vessel, electric heaters located outside the reactor vessel guard vessel were used to raise the temperature according to a stepwise temperature-rise pattern. While monitoring the temperature distribution in various areas during the heating process, it was demonstrated that the requisite temperature could be reached and maintained within 30 days.

In the preheating of the main cooling system and auxiliary system pipelines, the process was monitored to prevent the occurrence of any pronounced temperature unevenness inside the systems, and it was demonstrated that the requisite temperature could be reached and maintained within 14 days.

3.3 Sodium Charging and Initial Purification

After the systems were preheated to the requisite temperatures, sodium was loaded into the main cooling systems. The filling was done sequentially from the auxiliary system dump tank(s), by vacuum drawing through the system or by the operation of electromagnetic pumps. The operability both during the filling and drain operations, and the equipment and pipeline drain characteristics were verified.

Following this, purification system function tests were performed using the cold trap(s) in the sodium purification system. It was verified in these tests that the sodium in the systems could be purified to and maintained at adequately low oxygen concentrations, and the oxygen capture capabilities of the cold trap(s) were demonstrated.

3.4 Sodium Equipment Structural Integrity Verification

(1) Pipeline Heat Displacement Measurement Tests

Locations were selected in the main cooling system and auxiliary system pipelines in which pipeline heat displacement becomes large, and these displacements were measured, both during the preheating from room temperature to 200°C, and during sodium loading. The tests results agreed well with the analytical values for all systems, and were all within the range of values predicted during pipeline design analysis. Similar results were obtained in areas that were partially modified.

(2) Pipeline Vibration Tests

Representative pipe segments in the main cooling system pipelines were subjected to vibration tests after being charged with sodium. Based on the test results, characteristic vibration frequencies, vibration modes, and attenuation constants were analyzed, thereby verifying the propriety of the anti-earthquake design and integrity.

3.5 Measurement & Control Equipment Tests

(1) Testing of Instruments Inside Reactor Vessel

As part of the tests on the sodium liquid surface gauges that are peculiar to fast reactors, the liquid surface gauges inside the reactor vessel were zeroed and span-adjusted, using calibrated standard liquid surface gauges. Tests were also done to verify the effects of wave-suppressing tubes located around the liquid surface gauges, thereby verifying their effectiveness.

(2) Systematic Testing of Reactor Security Systems

In tests on the reactor security systems, a series of safety and security actions were confirmed to operate normally, in the prescribed sequences, from the detector terminal to the relay in the safety and security system equipment.

(3) Diesel Power Generating Equipment Function Tests

Start-up and load tests and load interrupt tests were conducted on the diesel power generating equipment that constitutes the emergency power equipment. It was confirmed that the plant can be operated according to the prescribed interlocks.

3.6 Operability & Controllability Verification Tests

(1) Primary Cooling System Sodium Flushing Operation

Using a temporary strainer (50 mesh) temporarily installed at the primary main cooling system stop valve, the system was subjected to a flushing operation with sodium (for approximately 50 hours at approximately 40% of the rated flow rate). It was verified that there are no abnormalities in the pump operation condition. Then, from the results of washing and visually inspecting the temporary strainer after it was removed at the completion of the flushing operation, it was confirmed that the cleanness controls implemented in the primary main cooling system (including the reactor vessel) are adequate.

(2) Primary Main Cooling System Circulation Pump Characteristic Tests

In May, 1992, as part of the primary main cooling system circulation pump characteristic tests, start-up and shutdown tests were performed, using the pony motors, together with trip tests, verifying that all of the operating characteristics are good. Also, the primary main cooling system of the Monju, which has three systems, is designed so that, even if two of the systems are shut down, the remaining system can remove the decay heat. To verify the design characteristics, the circulating flow volume was measured with two out of the three pony motors shut down. It was found that the necessary heat removal flow volume could be definitely achieved with a single motor.

Furthermore, from early July, 1992, start-up and shutdown tests using the main motor(s) were conducted, together with control system adjustment tests, thereby confirming that all three systems operate well at 100% flow volume. Also, main motor trip tests were conducted with the reactor trip circuit breaker "open." It was thereby verified from the flow volume coast-down characteristics both that the half-reduction time of the primary main cooling flow volume satisfies the design values, and that the relay to the pony motor operation operates smoothly.

In mid-July, 1992, the primary main cooling system circulation pumps were started up according to the start-up sequence, and continuous operating tests using the main motor(s) (for approximately 24 hours at 100% flow volume). During the operation, no abnormalities were found in the pump bearing temperatures or vibration levels, and it was verified that the primary main cooling system circulation pumps satisfy the function specifications.

(3) Secondary Main Cooling System Circulation Pump Characteristic Tests

Beginning at the end of April, 1992, the pony motors and main motor(s) for the secondary main cooling circulation pumps were sequentially started up, and start-up/shutdown tests and control system adjustment tests were conducted. The operating conditions of all three systems at 100% flow volume were found to be good.

Pump trip tests were also conducted, confirming that the flow-volume coast-down characteristics from 100% flow volume are according to design.

After running these characteristic tests on all the loops, beginning in August, continuous operation tests (for approximately 48 hours at 100% flow volume) were conducted on the loops in the secondary main cooling system, verifying that the secondary main circulation pumps satisfy the prescribed function specifications.

(4) Auxiliary Cooling Equipment Flow-Regulator Function Tests

With the smooth operation of the stop valve at the inlet to the secondary main cooling system steam generator and the stop valve at the outlet of the auxiliary cooling equipment air cooler, the sodium flow is diverted from the secondary main cooling system, so that it flows through the auxiliary cooling equipment. Tests confirmed that the requisite flow volume control systems operated normally.

Also, when the continuous operating tests were performed on the primary main cooling system circulation pumps, the heat removal volume of the auxiliary cooling equipment air coolers in the natural air-flow mode was measured with the sodium temperature at approximately 250°C, thus confirming that the heat removal capacity of the said air coolers roughly coincides with the values predicted by the analysis code. Furthermore, the heat removal capacity of the auxiliary cooling equipment was also measured at the stage of the performance tests which demonstrated adequate heat volume with the heat generated by the reactor core.

(5) Comprehensive Cooling System Tests

1) Cooling System Mode Operation Tests

The plant was operated with the primary main cooling system, secondary main cooling system, and part of the water-steam systems combined, in the basic operating modes that include the start-up/shutdown mode and low-temperature shutdown/fuel replacement mode according to the operating guidelines, thus verifying the propriety of the operating procedures. The water-steam systems were operated with hot water at approximately 195°C [illeg] and 127 kg/cm^2 x G from the auxiliary boiler passing to the evaporator, thus verifying the requisite operating time.

2) Auxiliary Cooling Equipment Simulated Start-Up Tests

The primary main cooling system, secondary main cooling system, and auxiliary cooling equipment were combined, and the reactor was manually shut down under conditions of 100% flow volume in the primary and secondary main cooling systems. It was thereby verified that the operation of the primary and secondary main circulation pumps switched to the pony motor drive, that the secondary sodium cooling flow path switched from the steam generator to the auxiliary cooling equipment air coolers, that the said air coolers went into an operating condition, and that the outlet sodium flow-volume and temperature control could be implemented smoothly.

Also, the cooling systems were raised to 325°C with pumped-in heat, with the primary and secondary main cooling system circulation pumps operating at 100% flow volume, thus permitting the collection of data such as the volume of heat dispersed from the cooling systems. The results of the cooling system temperature-raising tests are plotted in Figure 5.2.

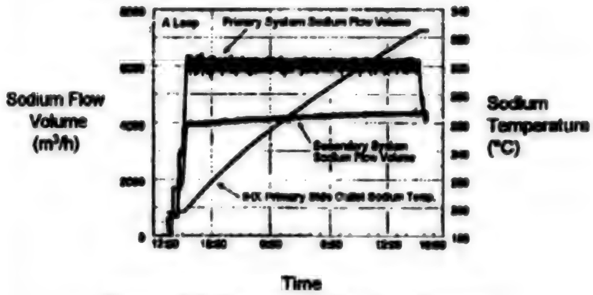


Figure 5.2 Results of Cooling System Temperature-Raising Tests

3) Primary (Secondary) Main Cooling System Circulation Pump Trip Simulation Tests

Under conditions of 100% flow volume in the primary and secondary main cooling systems, tests were conducted in which the shutdown of either the primary or secondary main cooling system circulation pumps were simulated. It was demonstrated that the signals from the reactor security system are issued in the proper sequence, that the primary and secondary main cooling system circulation pumps switch over to pony motor operation, that the auxiliary cooling equipment starts up automatically, and that the plant safely transitions to the shutdown condition (decay heat removal operating state).

4) External Power Supply Loss Simulation Tests

Tests were conducted which simulated loss of the external power supply, in the plant output operating condition and in the decay heat removal operating condition. These tests confirmed that the emergency diesel power generating equipment automatically starts up, that electricity is supplied according to the proper sequence, and that the plant can safely be transitioned to a shutdown state (decay heat removal operating state) using the emergency power supply.

3.7 Fuel Handling and Storage Equipment Function Tests

The reactor core configurational elements such as the fuel aggregates are handled separately by many pieces of equipment, including the fuel exchangers, fuel inserter-extractors, out-of-reactor fuel storage tank, fuel washing equipment, fuel canistering equipment, underwater trolley, fuel handler, underground trolley, and new fuel handler. Thus the function tests on the fuel handling and storage equipment is largely divided between the fuel replacement tests which are done when the reactor is shut down, and the fuel is exchanged between the reactor core and the out-of-reactor fuel storage tank, on the one hand, and the fuel transporting, processing, and storage tests which are done when the reactor is operating, and involves the transporting of new fuel to the out-of-reactor fuel storage tank, and the transporting of spent fuel, etc., from the out-of-reactor storage tank to the spent-fuel storage pool.

In the tests in room-temperature air, after the functions of each piece of equipment had been verified, the handling functions between the fuel inserter-extractors and the other equipment were verified by manual remote control and semiautomatic remote control from the main control and monitoring board for the fuel handling system. In particular, these tests, which were conducted while visually confirming the operating condition of the fuel handling equipment, and which included testing of core element handling by the fuel handling equipment in the reactor core and in the out-of-reactor fuel storage tank, address verification tests, and eccentricity tests, were significant in moving toward remote-control operations in sodium.

In the tests done in sodium, concerning the functions verified in room-temperature air, the effects of heat expansion and of the suspension forces caused by buoyancy in sodium were compared and verified, and the core elements, fuel inserter-extractor grippers, fuel replacement equipment, and in-reactor relay unit were washed to verify that the adhering sodium could be washed off. Also, it was verified that the combined operation of fuel replacement, fuel processing, and fuel storage could be done automatically from the main control and monitoring board for the fuel handling system, within the prescribed time.

High-precision and accurate movement is demanded in the fuel handling and storage equipment, due to its functions. Accordingly, in the comprehensive function tests, particular attention was given to the equipment shutdown precision (revolving action up and down action, travel and lateral movement, etc.), to action torque (including the suspension and aerial suspension loads for the core elements), and to operating time.

3.8 Simulated Core Configuration

In order to conduct the function tests on the fuel replacement equipment, control rod drive mechanism, and primary main cooling system equipment, dummy fuel aggregates, blanket fuel aggregates, and neutron shielding elements, totaling 716 core elements, were installed in the reactor core to configure a temporary (simulated) core. By executing this configuration of the simulated core in Monju, the propriety of the simulated core configuring procedures was confirmed.

In the tests on the control rod drive mechanism, normal drive tests were done, pulling out and reinserting the rods, and tests were also done in sodium, beginning with the control rods fully withdrawn and rapidly inserting them to simulate a scram operation, thus verifying that the requisite functions are exhibited during the sodium flow state.

3.9 Reactor Containment Vessel Leak Rate Tests

(1) Verification Prior to Sodium Charging

After installing the equipment, but prior to charging the system with sodium, reactor containment vessel leak rate tests were conducted. As a result, it was verified that the overall leakage rate of the reactor containment vessel prior to sodium charging fully satisfied the allowable leakage rate, and that the containment vessel exhibits the requisite airtightness. These tests provided basic data needed for the post-charging leak rate tests.

(2) Verification After Sodium Charging

After charging the systems with sodium, the containment vessel leak rate tests were repeated. As a result, it was verified that the overall leakage rate of the reactor containment vessel after sodium charging fully satisfied the allowable leakage rate, and that the requisite airtightness is exhibited even in the sodium-charged state.

3.10 Cell Liner Airtightness and Leak Rate Tests

In the unlikely event of a sodium leak, there are rooms (cells) where it is necessary to prevent the leaked sodium from coming into contact with air and building concrete. These cells are made completely airtight by lining them with steel liners. To verify the airtightness of these cells, leak rate tests were conducted in each cell located in the reactor building or the reactor auxiliary building. As a result, the leak rates for each cell liner was found to fully satisfy the allowable leak rate, and the requisite airtightness was exhibited.

3.11 Radioactive Waste Disposal Facility Function Tests

Function tests were conducted on the facilities for disposing of gaseous, liquid, and solid wastes.

In order to verify the hold-up performance of the activated charcoal absorption column in the gaseous waste disposal facility, the holding time of the activated charcoal absorption column was measured, using xenon and krypton gas, thereby verifying that the holding time is adequate.

In the liquid waste disposal facility, simulated waste liquid evaporation and concentration functions were verified in the waste liquid evaporation and concentration system, thereby confirming that the specified waste liquid can be disposed of.

In the solid waste disposal facility, simulated waste liquid was rendered into plastic solids, demonstrating that it can be disposed of in desirable solid form.

4. Fuel Transport and Reception

4.1 Fuel Transport

The fuel for the initial charge was transported to Monju in nine separate shipments. The fuel transported consisted of 205 fuel aggregates (including 5 test fuel aggregates) filled with plutonium-uranium mixed oxide fuel pellets, etc. Prior to the simulated core configuration, 177 blanket fuel aggregates were also transported.

The shipment containers used in transporting the core fuel aggregates were of a multi-ply structure, containing shielding and shock-absorbing material. These containers had passed tests in which they were dropped from a height of 9 meters, and fire tests in which they were heated to 800°C for 30 minutes, thus satisfying the national standards.

In transporting this fuel, a transport convoy was put together, complete with escort vehicles, and the operation was conducted very carefully, accompanied by radioactivity control specialists. Every safety precaution was taken, with the cooperation of local authorities all along the transport route.

The fuel was thus transported to the Monju facility according to plan and without any major obstacle or accident.

4.2 Fuel Reception

The receiving of the fuel aggregates at the site was accomplished in the following fashion.

When the fuel transport trucks reached the Monju plant, they were admitted one at a time to the fuel shipment area inside the reactor auxiliary building. The transport containers in which the fuel aggregates were packed were lowered from the truck beds with a special crane, and temporarily placed in a new fuel cask holding area. Subsequently, they were loaded onto new fuel cask transport trolleys (fuel container handling equipment) and, after inspection, sent to the new fuel handling room which is shielded.

The purpose of the new fuel inspection, after reception, is to insure the integrity of the fuel aggregates.

In the new fuel handling room, the covers were removed from the transport containers. Then the fuel aggregates were taken from the new fuel transporters, subjected to a reception inspection with fuel unit external inspection equipment, and then temporarily stored in the new fuel storage rack.

Next, in transporting the fuel aggregates from the new fuel storage facility to the out-of-reactor fuel storage tank, first the fuel aggregates are loaded into an underground trolley, according to the fuel transport plan. In the underground trolley, the fuel aggregates are preheated with heated gas and argon gas replacement is performed. Then the fuel aggregates are sent to a relay point between the underground trolley and the fuel inserter-extractor. The preheated fuel aggregates are transported by the fuel inserter-extractor to the out-of-reactor fuel storage tank that is filled with sodium, and stored there until the next fuel replacement operation is conducted.

The handling of the fuel aggregates is done by automatic remote-control from the fuel handling operating room. The flow of new fuel reception and handling equipment are diagrammed in Figure 5.3.

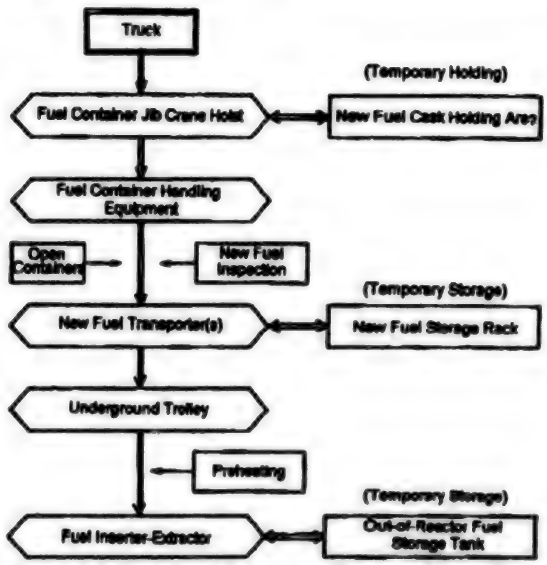


Figure 5.3 New Fuel Reception Flow and Handling Equipment

5. Performance Test Planning

At Monju, the tests that are performed to verify plant performance from reactor core fuel charging to normal operation begins are called performance tests.

5.1 Purpose, Category of Performance Tests

The main purposes of the performance test are to verify plant systems and equipment performance, in reactor output states from zero, after fuel charging, to the rated output, to evaluate the propriety of the designs based on the test data, and to collect hard data for future reactors.

During the performance test stage, test categories are selected, based on the following perspectives, and implemented.

(1) Verify plant systems and equipment functions and performance following core fuel charging

(2) Perform various adjustments to control systems and indicator instruments needed to run the plant

(3) Evaluate the design flexibility, operability, and maintainability, based on the performance test data

(4) Master the operating techniques required in FBR power plants

(5) Evaluate the results of Monju-related R&D done to date (such as the development of various sodium equipment, analysis codes, etc.)

(6) Acquire and collate hard data that will form the basis for developing future FBRs

Furthermore, in the Monju performance tests, the inspections stipulated by law were implemented, thereby verifying items related to function and performance.

The test categories planned according to the performance test purposes set forth above totaled approximately 130. These included both tests that were substantially the same as those implemented in light water reactor start-up testing, and tests done primarily for R&D purposes, to fulfill the role of an FBR prototype. The latter comprised roughly a third of the total tests.

5.2 Performance Test Schedule

The performance tests were divided amor ' plant characteristic preliminary tests, criticality tests, reactor physics tests, nuclear heating tests, and output tests. These are being conducted steadily, carefully, sequentially, and in stages, with thorough verifications and evaluations made of the results from each test.

The general schedule for the performance tests is represented in Table 5.3.

(1) Plant Characteristic Preliminary Tests

The heat-insertion that accompanies the primary and secondary main circulation pump operation heats the cold leg pipeline side of the primary and secondary main cooling systems, raising the temperatures up to the operating levels of approximately 397°C in the primary system and 325°C in the secondary system. Thus the plant operating characteristics are preliminarily evaluated from the thermal perspective.

(2) Criticality Tests

After the plant characteristic preliminary tests are completed, following the general inspection that is conducted, the criticality tests are begun.

Table 5.3 Performance Test Schedule

First, neutron sources are loaded into their proper positions in the blanket region. Then the dummy fuel aggregates that were temporarily loaded into the reactor core are sequentially replaced with core fuel aggregates until initial criticality is reached.

The replacement of the core fuel aggregates is divided into roughly nine steps. The number of fuel aggregates loaded in each step is determined by estimating the number of fuel aggregates required for criticality from the inverse multiplier curve.

After initial criticality is reached, preliminary measurements are made of the control rod reactivity values, the surrounding fuel reactivity values are measured, and other characteristics of the minimally critical reactor core are evaluated.

After that, the remaining core fuel aggregates are loaded into the core, thereby configuring the initial reactor core (198 aggregates). Following this, the excess reactivity of the initial reactor core is verified.

(3) Reactor Physics Tests

After the initial reactor core has been configured, reactor physics tests are conducted. These include tests to evaluate the control rod and fuel reactivity value characteristics, verify the reactor shutdown flexibility, and evaluate the shielding surrounding the reactor core, etc.

During the first half of the reactor physics tests, evaluations are made of the reactivity values of the various control rod aggregates, of the stationary absorber reactivity values, of the fuel reactivity and other reactivity value characteristics, and, using test aggregates loaded with radioactive foil, etc., of the in-reactor output distribution characteristics. In Figure 5.4 are depicted examples of reactor cores configured for measuring the in-reactor output distribution.

Following these tests, during the second half of the reactor physics tests, the outlet flow volumes are measured for core fuel aggregates that represent major regions, and the flow-volume distribution to the various core regions is evaluated.

In addition to flow-volume distribution evaluations, such reactivity coefficients as temperature coefficients and flow-volume coefficients are measured, as pertaining to the measurement of output coefficients in the output stage.

In the area of shielding characteristics, there are plans to evaluate the shielding surrounding the reactor, to evaluate the primary cooling system shielding, to evaluate the fuel handling system shielding, and to verify the air dosage equivalent ratios.

Concerning the radioactivity shielding evaluations, the neutron flux and radioactivity levels are evaluated, using radioactive foil and TLD, etc., in the

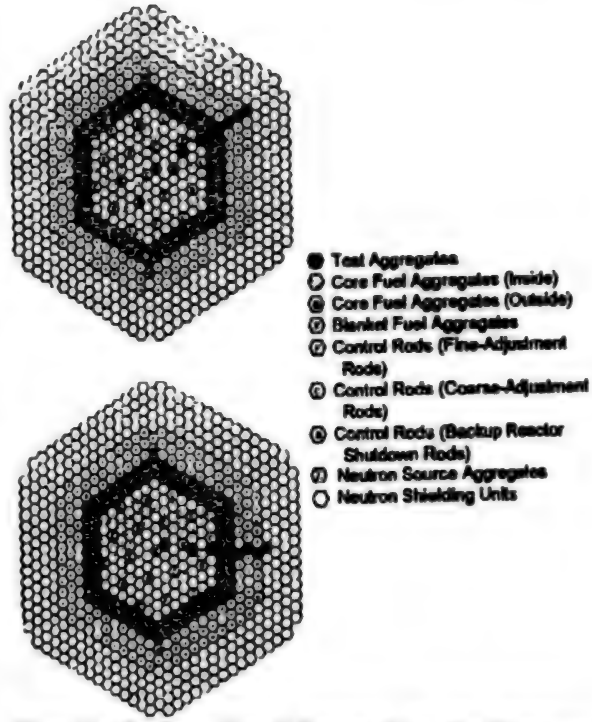


Figure 5.4 Configuration of Reactor Cores for Measurement of In-Reactor Output Distribution (Planned)

reactor containment rooms, and primary main cooling system rooms, etc.

The evaluation of shielding surrounding the reactor is to shielding evaluations of the reactor vessel upper plenum, the reactor vessel outer surface, and the shielding plug.

(4) Nuclear Heating Tests

Following the reactor physics tests, in preparation for the nuclear heating tests, equipment inspections are implemented. After that, the system temperatures are raised by nuclear heating, the integrity of the water-steam system equipment is verified under conditions of high-temperature, high-pressure water and steam, the turbines and generators are subjected to trial operations and adjustment, and preparations are made to make the first output connection to the electric power grid.

(5) Output Tests

After hooking into the electric power grid, power output tests will be conducted. The nuclear and thermal characteristics of the reactor core, and the radiation shielding performance, will be measured, in power output stages of approximately 40%, 75%, and full rated output. The operating characteristics, control characteristics, and design flexibility of the reactor equipment will be confirmed as well.

In confirming the plant characteristics when transitioning to each power output stage, the plant will be started up, operated, have its output varied, and shut down. The process quantities in each state will be measured.

Also, it will be confirmed that the reactor can be safely shut down, and the decay heat normally removed, even after a plant trip, generator load interruption, or external power supply loss.

After verifying performance at each output stage, as the final performance test, the plant will be continuously operated at rated output to confirm that the plant systems and equipment can be so operated, stably and without malfunction.

By implementing the tests described above, all of the performance and function verifications needed prior to Monju's going fully on line will have been accomplished.

6. Reaching Criticality

Let us summarize events up to the achievement of initial criticality.

Prior to configuring the critical reactor core, two neutron source aggregates (^{252}Cf) were loaded into the core. In addition, when configuring the simulated reactor core during the comprehensive function tests, the core was charged with blanket fuel aggregates, neutron shielding units, and control rod aggregates.

The required number of core fuel aggregates were moved to the out-of-reactor fuel storage tank (EVST, the place where new fuel is stored temporarily prior to loading it into the reactor). These fuel aggregates were then moved out of the EVST, one at a time, and transferred to the fuel inserter-extractor. After that, they were carried inside the reactor containment vessel on a truck running on rails which link the EVST and the top of the reactor. The core fuel aggregates were then inserted into the core from the top of the in-reactor relay unit which forms the boundary with the reactor vessel, and held in a revolving rack at the bottom of the said unit. In parallel with this operation, inside the reactor vessel, the dummy fuel aggregates (having the same outer form as the real ones, but containing no nuclear material) that were used to configure the simulated reactor core for the comprehensive function tests were drawn out of the core by the fuel exchanger positioned at the top of the reactor vessel, moved, and secured in another revolving rack of the in-reactor relay unit. In this manner the core fuel aggregates and the dummy fuel aggregates were exchanged by the revolving mechanism in the in-reactor relay unit.

The Monju reactor core fuel regions are divided, forming two reactor cores of uniform region, with the high-plutonium-enriched core fuel aggregates on the outside, and the low-plutonium-enriched core fuel aggregates on the inside. For this reason, both the reactor core structure and the aggregates themselves must be made to prevent the erroneous charging of these fuel aggregates having differing plutonium enrichment. The core fuel aggregates in the revolving rack in the in-reactor relay unit after exchange with the dummy fuel aggregates were loaded into their core positions after the dummy fuel aggregates were pulled out by the turning and vertical motions of the fuel replacement equipment and the turning of the revolving plug. The core charging of the core fuel aggregates was performed with the sodium liquid level inside the reactor vessel at the normal level, the sodium temperature at approximately 200°C, and the sodium flow volume held at 10% of the rated flow volume.

Meanwhile, the dummy fuel aggregates were moved via the in-reactor relay unit and fuel insertion-extraction equipment to the out-of-reactor fuel storage tank and kept there temporarily.

The charging of the core fuel was performed in two stages, namely the charging of the inner core region and the charging of the outer core region. The 108 fuel aggregates in the inner core were charged in clockwise fashion from the center of the core, from October to November, 1993. The fuel aggregates in the outer core began to be charged in January, 1994.

After the requisite amount of fuel had been charged (approximately half the number of fuel aggregates predicted to be needed for minimal criticality), the control rods were pulled out. When this was done, the sodium in the reactor vessel was at a temperature of approximately 200°C, and the flow volume of primary coolant was approximately 49% of the rated flow volume. The reactor control system is made up of a main reactor shutdown system (fine-adjustment and coarse-adjustment rods) and a backup reactor shutdown system (backup reactor shutdown rods). In retracting these control rods, as when the reactor is operating normally, all of the backup reactor shutdown rods were pulled out, and then, while monitoring the change in neutron count, the fine-adjustment rods and coarse-adjustment rods were pulled out sequentially. From the neutron count rate obtained with the retraction of the control rods, the inverse multiplier curve was determined, the number of remaining fuel aggregates needed to be charged to reach the critical state was estimated, and roughly half that number was loaded into the reactor. In this manner was the approach to criticality made.

This series of operations, namely the fuel charging, measurement of neutron count rates, and estimate of the number of fuel aggregates required for minimal criticality, was implemented repeatedly, based on the test results, and initial criticality was attained at 10:01 a.m. on 5 April 1994. The number of fuel aggregates charged in the core at the time of initial criticality was 168. The operation is summarized in Figure 5.5. Criticality was achieved with all of the control rods out of the center of the core retracted, and confirmed with the multiplier curve of the neutron count rate. The position of control

rod retraction from the core center at the time of criticality was approximately 730 mm (stroke approximately 1000 mm). The core configuration with the 168 core fuel aggregates at the time of criticality is diagrammed in Figure 5.6.

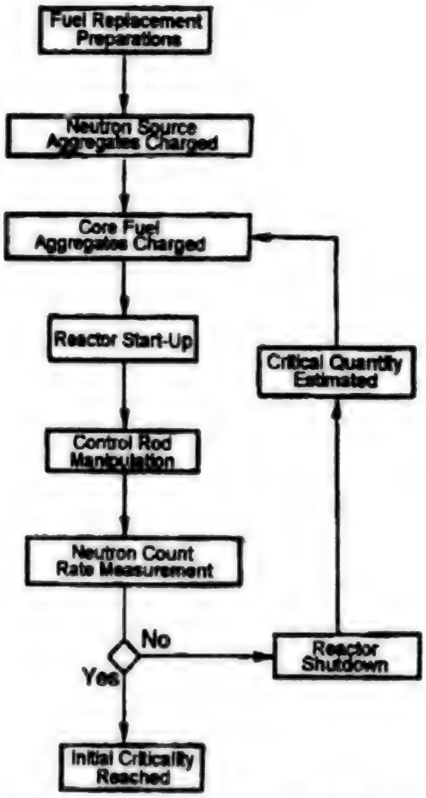
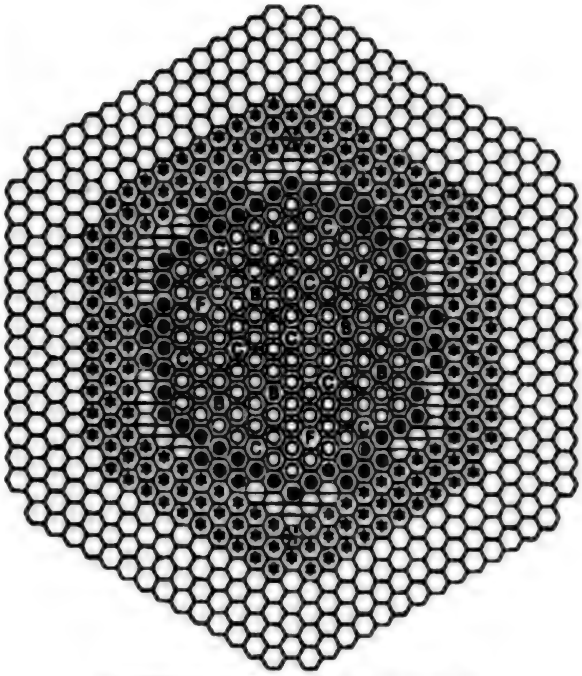


Figure 5.5 Summary of Critical Core Configuration Operations



Reactor Core Configurational Elements		Symbol	Quantity
Reactor Core Fuel Aggregates	Inner Core		108
	Outer Core		60
Dummy Fuel Aggregates			30
Blanket Fuel Aggregates			172
Control Rod Aggregates	Fine Adjustment Rods	F	3
	Coarse Adjustment Rods	C	10
	Backup Reactor Shut-Down Rods	E	6
Neutron Source Aggregates			2
Neutron Shielding			316
Surveillance Aggregates			8

Figure 5.6 Minimal Critical Core Configuration

Based on the actual fuel composition, three-dimensional dispersion nucleus computations (Hex-Z) were performed, and, from the results, after taking mesh, transportation, non-homogeneity, and other correction factors into consideration, the effective multiplication factor estimated for the 168-element core was 0.9995 +/-0.0045, thereby verifying the propriety of the prediction and evaluation methods.

7. Future Directions

Following the attainment of initial criticality, after configuring the initial reactor core with all of the 198 fuel aggregates of the initially charged core, performance tests were conducted, namely reactor physics tests, nuclear heating tests, and output tests.

The initial entry into the power grid—the next major milestone in the performance tests—will be conducted in April, 1995. In December, 1995, with continuous load tests at full rated output, all of the prescribed tests in the performance tests conducted over a 3-year period will have been completed. At that time the plant is scheduled to enter the final stage of full-scale operation.

Advancement of FR Development and Monju Evolution

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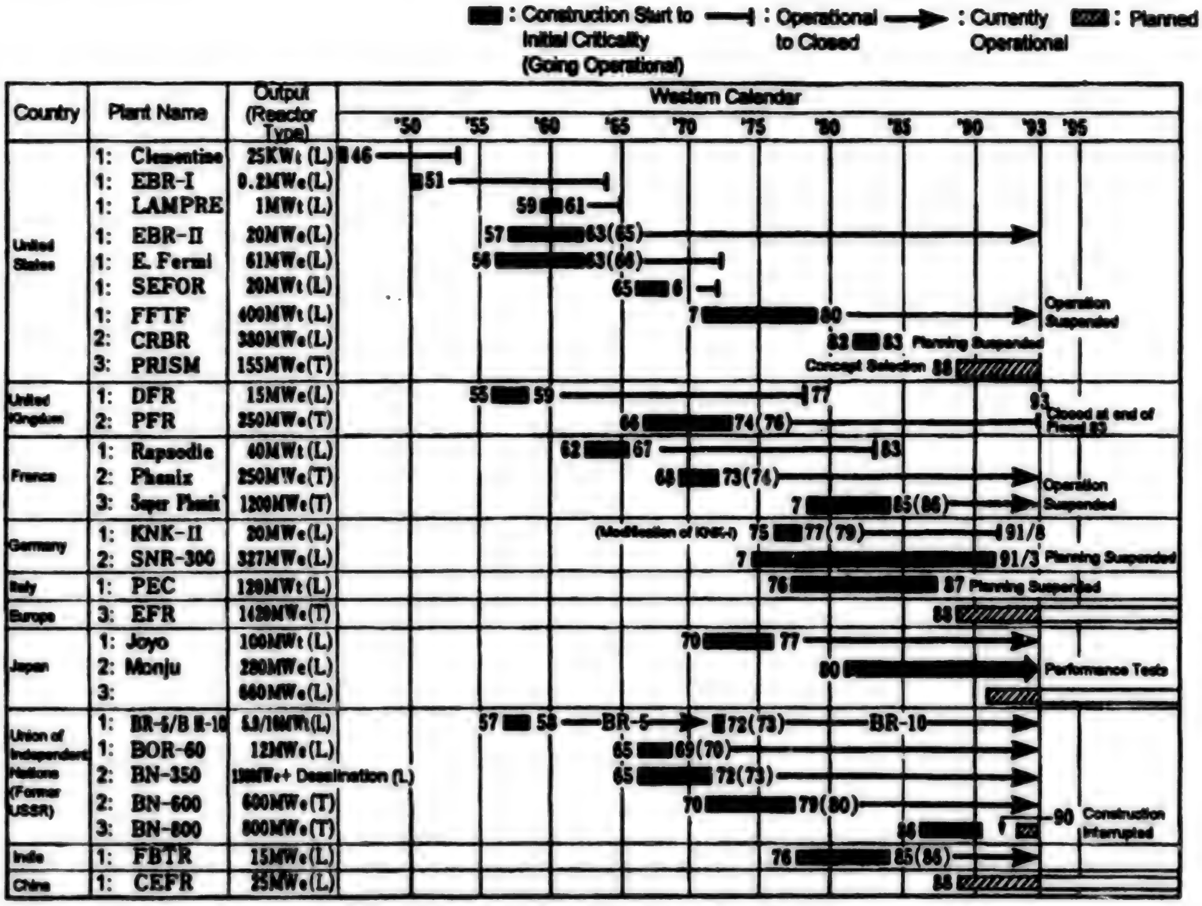
[FBIS Translated Text]

1. World Fast-Reactor History & Development

The leading nuclear powers in the world have for a long time been making great efforts to practically implement fast reactors, conscious that these will become the main pillar of atomic energy in the future.

This development has for the most part followed a common pattern, through the steps of experimental



1: *Jikken*: Experimental Reactor; 2: *Gen*: Prototype Reactor; 3: *Sho*: Demonstration Reactor; 4: L: Loop Type; 5: T: Tank Type

Table 6.1 World Fast-Reactor Development History

reactor, prototype reactor, demonstration reactor, and on toward practical reactors. In Table 6.1 the world history of fast-reactor development is diagrammed.

We now give an overview of fast-reactor development history and future planning in various countries.

1.1 [Passage omitted]

1.2 United Kingdom

The United Kingdom began developing fast reactors in 1951, and moved with relative rapidity to the prototype reactor stage.

Achieving criticality in 1959 with the NaK-cooled experimental reactor DFR, valuable information on fuel technology was provided, but the plant was closed in 1977.

The prototype reactor PFR went critical in 1974, but was plagued with continual leaks from the steam generator from early on, demanding repair and modification. Subsequently, trouble was again experienced with the steam generator in 1987. Excluding that period, the plant was operated steadily until June, 1991. After that, it was shut down due to problems with the primary cooling system. After evaluating the effects of this, and possible corrective measures, operation has continued since March, 1993. After continuous operation until March, 1994, the PFR plant was shut down for closing.

As to development since the prototype reactor, the British have participated in the European fast-reactor project (hereinafter "EFR project"), but announced the cessation of government funding for this project in November, 1992.

1.3 France

France began developing the fast reactor in 1957. Development work has been done mainly by the Atomic Energy Agency (CEA), with the focus on sodium cooling and oxide fuels.

The experimental reactor Rapsodie achieved criticality in 1967, was operated until 1983 for irradiating fuels and materials, and then shut down. It is now being dismantled.

In 1973 the prototype reactor Phenix went critical, and was operated until 1990 for the purposes of verifying the prototype's operating characteristics, obtaining reliability and economical data, and irradiating fuels and materials. In 1989 and 1990, however, mishaps occurred which lowered the reactivity, and operation is now being conducted to determine the cause.

The demonstration reactor Super Phenix became critical in 1985 and achieved 100% output in 1986. However, in March, 1987, sodium leaks were discovered in the out-of-reactor fuel storage tank for spent fuel and operation was stopped. Operation was resumed in January, 1989, but air leaked into the primary system sodium in July, 1990, followed by a partial cave-in of the turbine building roof in December of that year due to an unprecedented heavy snowfall, and operation was stopped. The trouble with the spent-fuel storage tank was handled by not using it for storing spent fuel, but rather designing it for use as a relay station for getting fuel into and out of the reactor. The material was changed and it was newly fabricated and converted. During this time, in order to gain approval to resume operation, thoroughgoing studies were made of the operating regulations and measures against sodium leaks in the secondary system, but the resumption of operation was delayed by order of the prime minister (June, 1992) after responding to questions from the Atomic Energy Facility Safety Bureau, which examined the changes. Subsequently, the measures against secondary system sodium leaks were beefed up, whereupon another petition for a permit to resume operations was made. Procedures are currently being followed for the resumption of operations, holding public safety hearings, and submitting a report on the results of safety investigations made by the Nuclear Reactor Facility Safety Bureau.

In February, 1994, the role of the Super Phenix was changed from power reactor to research reactor, and the prime minister issued a communique announcing that the reactor would be used to demonstrate technology.

As to development work subsequent to the Super Phenix, the EFR project is now being promoted.

1.4 Germany

Germany (formerly West Germany) has been involved in fast-reactor development since 1960.

The hot neutron reactor KNK-I was modified into a fast reactor, and, in 1977, the experimental reactor KNK-II began operating. These facilities are used for experimentation in a wide range of fields, including irradiation technology, fuel cycle research, and instrumentation technology. The facilities were shut down in August, 1991, having completed their mission.

The fast reactor SNR-300, after construction was completed, failed to obtain a permit from the state government for fuel transport and storage, and so trial operations could not be conducted. Then, with the worsening of the economy due to German reunification, the project was suspended in 1991.

As to development work since the demonstration reactor, Germany has been participating in the EFR project. However, as of the end of 1993, related R&D has been suspended.

1.5 European Joint Fast Reactor Development Project (EFR Project)

In 1984, the European nations of the United Kingdom, France, West Germany, Italy, and Belgium signed a memorandum concerning the development of the European Fast Reactor (EFR), and proceeded with development work. This is to prepare for the demand for commercial fast reactors which should emerge around 2010 when current light water reactors will be rebuilt. The idea is to consolidate all the building and operating experience, and the design of demonstration reactors, which has been accumulated separately by the various nations, and work together to achieve greater efficiency and economy.

The conceptual design work in this project has been finished, with the objective being to demonstrate the economies to be gained by shifting from the current light water reactors, and to have permits and approvals from permits to be accepted by the participating countries. Over a 3-year period, beginning in April, 1990, studies have been done to verify the propriety of the conceptual design, and other R&D has been conducted.

As to subsequent plans, the word is that design research and basic R&D will be pursued, but on a smaller scale due to the cessation of funding by the United Kingdom.

1.6 Union of Independent Nations (Former Soviet Union)

In the Union of Independent Nations (the former Soviet Union), development work on fast reactors began in 1949. In 1955 the plutonium-charged critical aggregate BR-1 became critical. This was followed by the construction of BR-2, BR-3, and other facilities. Based on the tests done with these facilities, the decision was made to proceed with the development of sodium cooling and ceramic fuel. With this approach, the experimental reactor BR-5 (critical in 1958), BR-10 (modification of BR-5; critical in 1972), BOR-60 (critical in 1969), and BN-350 (critical in 1972; used for both electric power and desalinization) were built. BN-350 experienced steam generator leakage problems from 1973 to 1975, but has been running steadily since being repaired.

The BN-600 prototype reactor which followed reflected the experience of operating the BN-350 and other reactors. This reactor became operational in 1980. From 1983, this plant was operated with a facility utilization rate exceeding an average of 70%.

Turning to the development of demonstration reactors, it was originally planned that four BN-800 plants would be built. Construction work on two of these, in Beloyarsk and the South Urals, got underway in 1986, but the work was suspended temporarily in 1990.

In December, 1992, the Russian government issued a proclamation entitled "Problems With Construction of Nuclear Power Plants Inside Russian Federal Territory." This proclamation contained plans to build four BN-800 fast reactors, with two of these to become operational around the year 2000.

1.7 India

The fast-reactor development policy in India was adopted in the later 1960's.

According to the Indian policy, the experimental reactor FBTR would be designed and built, in cooperation with the French Atomic Energy Agency (CEA). Criticality was achieved in a small reactor core in 1985. Subsequently, after problems with fuel replacement in 1987, function tests were conducted with a heat output of 1 MW. In the future, there are plans to change to a normal reactor core configuration and move toward rated output operation.

Preliminary design work is being done on the prototype reactor PFBR.

1.8 China

China began doing basic research on fast reactors in the late 1960's. Based on the results of this basic research, the Chinese decided on a development strategy that envisions experimental reactors, modular fast reactors, and large fast reactors. The first step will be the experimental reactor CEFR (25 MWe). Design work is being done with hopes of construction by 2000.

2. Contributions of Joyo

Through the design, construction, and operation of Joyo, technology was accumulated in fields such as fast-reactor physics, reactor control, sodium technology, and fuel technology.

Some representative examples are cited below.

(1) It was seen that it would be possible to stably configure a high-output-density, high-speed neutron reactor by combining plutonium-uranium mixed oxide fuel, sodium cooling, and stainless steel structural materials.

(2) It was seen that the reactor could be controlled by an out-of-reactor neutron counter system and a control rod drive mechanism mounted atop the reactor.

(3) The technical feasibility of handling sodium coolant and cover gases (including mists and steam), and of handling heat transfer flows in the sodium equipment, was realized.

(4) The feasibility of fuel replacement below the shielding plug was realized.

Through the accumulation of such experience, the technical experience needed to build a prototype reactor was gained.

The purpose of the Monju plant, which was the second step in developing fast reactors, was to practically implement fast-reactor technology in Japan, by developing, without outside help, a prototype that would lead to demonstration and practical reactors. Through the experience of designing, fabricating, building, and operating Monju, the primary goal would be to demonstrate the viability of the fast-reactor power plant concept in terms of initial performance, safety, reliability, and operability.

The main plant concepts for Monju were put together on the basis of the former experience in developing the Joyo plant, and in developing the sodium heated steam generator used in large experimental facilities in steam supply systems.

The scale expansion factor in the core heat output of the former is approximately 7 (100 MW $\rightarrow$ 714 MW), while that of the heat exchange volume per loop in the later is approximately 5 (50 MW $\rightarrow$ 238 MW).

In terms of Monju plant performance, and in view of the purposes stated above, the approach would be to pursue the performance expected from a fast reactor in terms of multiple objectives. Thus a number of design levels were established, including the plant heat efficiency (approximately 40%), average core output density (approximately 280 kW/l), average fuel burn rate (approximately 80,000 MWd/t), and breeding ratio (approximately 1.2).

In conducting the integrated equipment design and safety design for Monju, design policy and design analysis programs were devised, reflecting related R&D gains and the Joyo construction and operation experience, and design work was conducted accordingly. This development experience, combined with the gains in related basic technology development, will provide a solid foundation upon which to establish design methods for future fast-reactor power plants. In this manner the various equipment and systems were designed and Monju was constructed.

Construction management includes safety management, quality control, and process management. Quality control efforts were coordinated, duly respecting the autonomy of each company involved, but seeking as much as possible to promote a unified policy, and giving particular attention, in view of the nature of a sodium reactor, to cleanness control, contamination prevention, and salt incursion prevention.

The cleanness control criteria and the control criteria used in preventing contamination by foreign matter when building the Joyo plant were reviewed to facilitate the enforcement of even stricter controls.

The purpose of the trial operations were to make the finished prototype plant very safe and reliable, and to demonstrate and develop technologies leading to a demonstration reactor. These trials were done in two stages, namely comprehensive function tests and performance tests. These were conducted to comprehensively verify the system functions throughout the plant. These tests

were implemented in the context of the three test phases established for Joyo, namely in room-temperature air, in argon gas, and in sodium.

In selecting the individual test categories, the selections were made from a wide range of test category candidates, for three purposes, namely to

(1) verify the design performance

(2) verify the operational performance, and

(3) obtain data for further R&D.

A total of approximately 300 tests were conducted, including approximately 240 tests peculiar to sodium-cooled fast reactors.

The purpose of the tests done in room-temperature air was to verify the equipment functions and performance prior to sodium charging. Particular care was taken during these tests to verify the performance and functions of the fuel handling equipment and control rod drive equipment, system which it would be impossible to visually inspect after sodium charging. In the argon gas tests, the subjects were the reactor vessel and such sodium systems as the primary and secondary cooling systems. The atmosphere inside these systems was converted from air to argon gas. Then they were preheated with electric heaters to approximately 200°C. Then the system airtightness, equipment operation, and displacement, etc., were verified in the high-temperature condition. The preheating performance of the pipelines and large equipment was verified prior to sodium charging. It was also confirmed that, with conversion to argon gas, low oxygen concentrations were achieved inside the systems, and that all of the systems were very leak-tight.

Then, in the tests done in sodium, after the systems were charged with sodium, the functioning of each sodium pump was verified, and then tests could be conducted in progressive steps to verify the relative system interactions and plant performance in the primary, secondary, and water-steam systems.

Finally, comprehensive cooling system tests were conducted to verify the normal operating mode performance of the overall plant in a configuration close to that of actual operation, following the basic operation plan, and to verify the responsiveness to transients such as main circulation pump trip events, etc.

Through these series of tests, it was possible to confirm that the system equipment configuring the Monju plant exhibits the proper system functions and performance, and to establish the conditions under which it would be possible to charge the reactor core with fuel. Furthermore, Monju design propriety evaluations, together with design flexibility analyses and design code verifications were made, for the purpose of bringing the Monju project technology together, and to collate the initial plant data obtained from the comprehensive function test results.

We wish now to cite two or three examples of how the gains made with Joyo were reflected during the comprehensive function tests at Monju.

The configuration of the simulated reactor core, which was the first item in the tests done in air at room temperature, was done for the following purposes.

(1) To form the flow paths for the primary main cooling system, inclusive of the reactor core, by using dummy fuel and neutron source aggregates, thereby making it possible to test the operating performance of the primary main cooling system circulation pumps

(2) To facilitate in-reactor tests of the control rod drive mechanism and fuel handling equipment

Each of the core configuring elements used in Joyo measured 2950 mm and weighed about 60 kg, so that two people could manage to lift it. The Monju elements, however, are approximately 4200 mm long and weight 200 kg each. Furthermore, the space inside the Monju reactor is much larger than the Joyo reactor, the inner diameter of the trunk being 7.1 m compared to 3.6 m. Thus a temporarily installed crane was carefully used to load each element into the reactor core.

The procedure for configuring the simulated reactor core was as follows. Placing great emphasis on improving the installation precision of the core elements, and drawing from the Joyo experience, in the initial installation phase the core elements were inserted in order, beginning at the periphery and working toward the center to form a framework. The reactor core was divided into three regions. Then the remaining elements were installed from the outside in, in each region. Moreover, this work is done in a work environment that is narrow and dark, demanding great care in terms of work safety, foreign matter, and cleanness control. The Joyo experience again proved invaluable during this operation.

After thoroughly verifying the functions of the control rod drive mechanism and fuel handling equipment in the room-temperature air tests, beginning in September, 1991, sodium was loaded into the systems and the reactor vessel. In this operation, the air that was in the reactor vessel and the primary cooling system, etc., was exchanged for high-purity argon gas, and preheating tests were begun in which these systems were gradually raised to the target temperature of 200°C with electric heaters.

In planning the preheating tests, the experience gained in similar operations with the Joyo reactor were fully exploited. In Joyo, the outside of the reactor vessel is a leak-jacket structure, so high-temperature nitrogen gas was circulated through it with blowers. However, with this gas heating method, the amount of dissipated heat became enormous, the temperature differentials in the circumferential direction became large due to natural convection, and temperature control was extremely problematic.

The Monju reactor vessel is a large tank which has an inner trunk diameter of approximately 7.1 m and an

overall length of 18 m. It also has mechanisms atop the reactor as well as an in-reactor structure. Therefore, intense studies were done from the design stage, drawing from the Joyo experience, with a view to devising a way of raising the temperature of the reactor vessel and the inside of the reactor uniformally. In order to avoid the circumferential and vertical temperature differentials due to natural convection in the internal argon gas, preheating tests were conducted on a 1/3-scale model, the necessary heat transfer data for preheating analysis were collected, and detailed analyses were conducted.

The preheating was done with electric preheaters. In the Monju preheating tests, it proved possible to limit the circumferential temperature differentials in the reactor vessel, making the preheating as uniform as possible, and to establish radial temperature differential limits for the in-reactor support structure in the reactor vessel, thereby achieving the preheating in the prescribed timeframe while keeping within the said limits and obtaining results that coincided well with the before-the-fact analyses.

In these tests, the natural convection prevention mechanisms were verified against the analysis results, to a high degree of precision, and they proved to be a major factor in obtaining good test results. The natural convection phenomenon was studied at the Oarai Engineering Center, and the results implemented in Monju. For example, the narrow gap between the stationary plug and the reactor vessel was fitted with convection prevention plates around the entire circumference, drawing from the Joyo preheating experience. Anti-convection baffle plates were also installed in the gap between the inner and outer casings, all around the circumference, in the primary circulation pump(s).

In the tests done in sodium, sodium purity control was a major element. Oxygen, hydrogen, and other contaminants adhering to the surfaces of the system equipment dissolve into the sodium loaded into the systems, impairing the purity of the sodium. In order to get this back to a high purity level, sodium purifying equipment is installed in the sodium systems.

Two methods are used concurrently to control sodium purity. One is the use of plugging meters built on-line in the systems. The other is the direct sampling and chemical analysis of the sodium flowing through the systems. Based on these results, the cooling temperature of the cold trap is regulated, and the oxygen contained in the sodium thereby captured.

As to the cold trap temperature control, if the set temperature is precipitously lowered, clogging occurs early, while if the set temperature is lowered gradually, the purification operation becomes too long. From the Joyo experience, optimum purification conditions were determined for Monju. Specifically, purification is done with the plug temperature—cold trap set temperature ≤ 5°C, in a period of 50 days in the primary system. In this way, purification of from 9 ppm to 3 ppm, in terms of oxygen concentration, was accomplished according to plan.

The Joyo experience is also reflected in the structure of the cold trap. The original Joyo cold trap was a bottom flow-in type which exhibited a tendency toward local clogging. To overcome this problem, the structure was changed to a mesh total-surface flow-in type in which a bottom-open thermally insulative gas layer is provided so that there is no heat exchange between the sodium on the high-temperature side and that on the low-temperature side. The capturing mesh packing ratio is lowered, and the cold trap itself is made more long-lasting.

In the Monju, building on these experiences with the Joyo reactor, a cooling pipe built-in approach is taken in the sodium cooling systems in the interest of better cooling effectiveness. The total-side-surface flow-in method is used in leading the sodium to the mesh in order to widen the effective impurity capture area. In addition, the insulative gas layer structure is adopted to control the heat in-flow to the cooled sodium.

These are just a few examples of how Joyo technology is reflected in the Monju plant.

For 16 years since it became critical, the Joyo reactor has been operated with great success. During this time, many modifications and improvements have been made in tests, operations, maintenance, and repair. The experience gained and the improvements made in the reactor fuel, control rods, neutron counters, failed fuel aggregate detection system, reactor vessel longitudinal liquid level meter, cover gas on-line gamma-ray monitor, equipment malfunction monitoring system, pipeline support systems, exhaust gas equipment, waste liquid processing facilities, secondary system preheater control system, secondary purification system control, operation and maintenance support systems, primary and secondary spent-fuel storage facilities, irradiated fuel assembly inspection facilities, and operational training simulator, etc., have led to definite advances in the development of the fast reactor.

At the Joyo facility, the MK-III project is being promoted, and studies are being conducted on a demonstration project for revolutionary new element technology, involving expanding current capacity, namely to 1.4 times the current heat output, approximately 1.3 times the current high-speed nuclear flux, and approximately three times the high-speed neutron irradiation space. In the MK-III project, in addition to responding to the demand for irradiation applications in wide high-speed neutron flux sites, including fuel rupture limits and transients, and over-output tests, in the project for demonstrating revolutionary new elements, there are plans to work on demonstrating a new type of reactor shutdown mechanism, and an intermediate heat exchanger which uses SUS316 FR material. By having power generating equipment, and scaling up the heat output roughly 7 times that of the Joyo reactor, the plant would have a scale and configuration closer to that of a demonstration reactor or full-scale reactor, and be used to develop and demonstrate various performance factors as such. In other words, together with Monju, which was

built to develop and demonstrate fast-reactor technology, that is, reliable and extremely safe fast-reactor power plant technology and improved economies for future reactors, Joyo will continue to play an important role in Japan's development of fast-reactor technology.

Each thing we learn becomes a gain, a success, and a technology that can be exploited in the next generation of reactors. In the course of developing fast reactors, from experimental to prototype to demonstration and on to full-scale operational reactors, our people will be gaining experience, obtaining new knowledge, doing research and development work, and advancing on to higher levels of technology. In this context, Joyo will perform as a pivotal and basic entity in Japanese fast-reactor development, and the technology it gives birth to will contribute enormously to the future.

3. Monju Influence on Demonstration Reactor

We now discuss how the experience and successes garnered in the design, fabrication, installation (site construction work), and operation of the Monju reactor will be reflected in the development and construction of the demonstration reactor and on into the future. For this section we have referred to the "Preliminary Conceptual Design" which embodies the design research done in the field of electric power demonstration reactor research up through 1991 at the Japan Atomic Power Company (JAPC). The specifications in the preliminary conceptual design are summarized in Table 6.2. A conceptual diagram of the reactor building and reactor structure are given in Figures 6.1 and 6.2, respectively.

Table 6.2. Specification Comparisons Between Monju and Demonstration Reactor Preliminary Conceptual Design

Basic/Major Specification	Monju	Demonstration Reactor Prelim. Conceptual Design
Reactor type	Trunk flow-in/flow-out loop type reactor	Top-entry loop type reactor
Heat output	714 MW	1600 MW
Number of loops	3	3
Reactor outlet temperature	529°C	550°C
Main steam temp./pressure	483°C/127 kg/cm^{2}G	495°C/169 kg/cm^{2}G
Reactor core type	Homogeneous core	Homogeneous core
Fuel	MOX	MOX
Average burn rate	Approx. 80,000 MWd/t	Approx. 90,000 MWd/t* Approx. 150,000 MWd/t**
Breeding ratio	Approx. 1.2	1.05-1.2
Reactor wall protective structure	Two liquid levels at start-up	Low-temperature Na circulation
Reactor vessel material	SUSF304	SUS316
Decay heat removal system	Secondary system branching (IRACS)	Direct core cooling system (DRACS)
Steam generator	Flow-through helical coil-branched type	Integrated flow-through helical coil type
Heat transfer pipe material	2 x 1/4Cr-1Mo steel/SUS321	Improved 9Cr-1Mo steel
Spent fuel decay-holding storage method	Out-of-reactor storage	In-reactor storage

3.1 Basic Design

In the design work that determines the basic plant and equipment specifications (that would insure the economy and safety levels demanded in a demonstration reactor, and open the way to full design), the Monju experience will be widely reflected. In designing the reactor core, based on the results of the Monju performance tests, reactor analysis techniques will be improved and developed, nuclear constants will be revised and organized, and this will be reflected in the demonstration reactor. In the shielding design, the Monju experiences in terms of streaming measures for the shielding around the reactor and the cooling system equipment, and the shielding design techniques that were brushed up on using the results of the performance tests, will be reflected in the demonstration reactor. In the area of core flow-volume distribution, the techniques established with Joyo and Monju will probably be reflected, while it is believed that, with the Monju results, high-precision analysis can be made of flow-volume distribution errors. Concerning heat flow dynamics design, phenomena peculiar to sodium reactors were elucidated during the Monju design phase, after which a number of codes were developed, including the 3D in-liquid heat flow analysis code AQUA, the 3D in-gas heat flow analysis code FLUSH, and the system dynamic characteristic analysis code Super-COPD. These are powerful tools for designing the demonstration reactor. Liquid surface behavior is an important problem for circulation systems, including equipment containing a liquid surface. Wide-ranging case studies are being made with Monju, which will be helpful in the demonstration reactor design. Concerning system design work in which probability safety analysis (PSA) is used, applications to the demonstration reactor will be based on the Monju experience. Concerning measures to deal

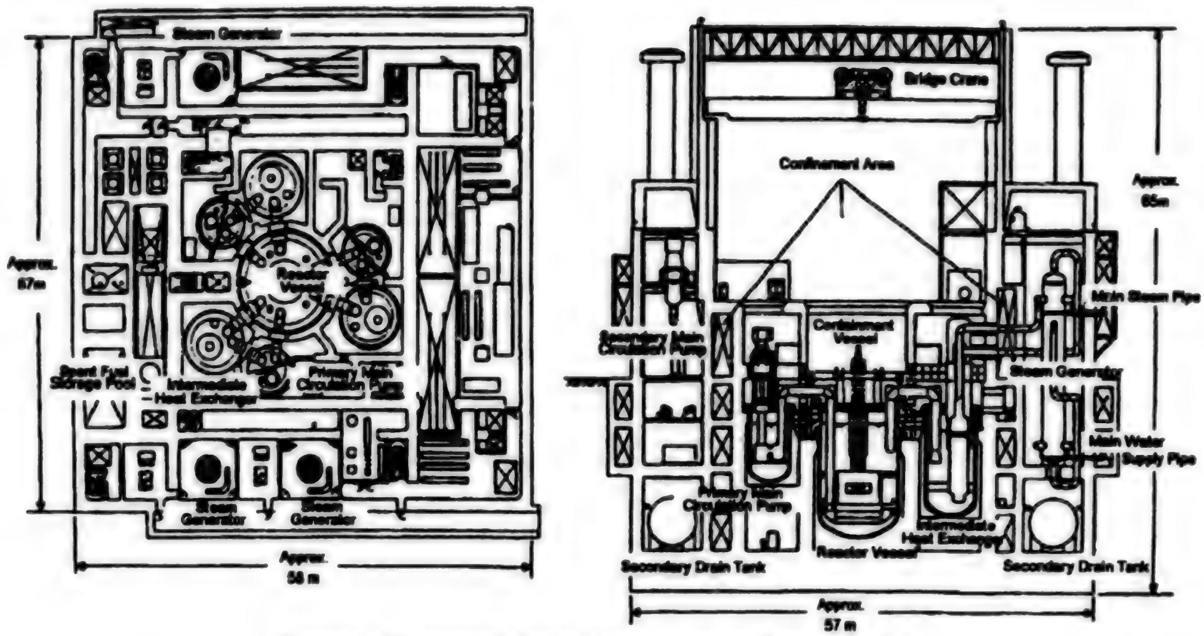


Figure 6.1 Conceptual View of Demonstration Reactor Building

with sodium fires, detailed studies, including R&D, are being conducted with Monju, and, according to the degree of demand for the demonstration reactor, design procedures and means can be provided. Problems involved in up-scaling were experienced in building both Joyo and Monju, and these experiences will be reflected in the demonstration reactor. Concerning airborne radiation dosage evaluations, the analysis methods used for Monju will be brushed up on using trial operations and operation results, and then used in analyses for the demonstration reactor.

3.2 Equipment Design

The grounds for approving the detailed design and fabrication design of the equipment in general include:

(1) that which has proven successful,

(2) that which has been improved through the course of operating it,

(3) that which, having no basis in (1) or (2), has been verified through R&D,

(4) that which, as the next best option, involves the partial application of the foregoing.

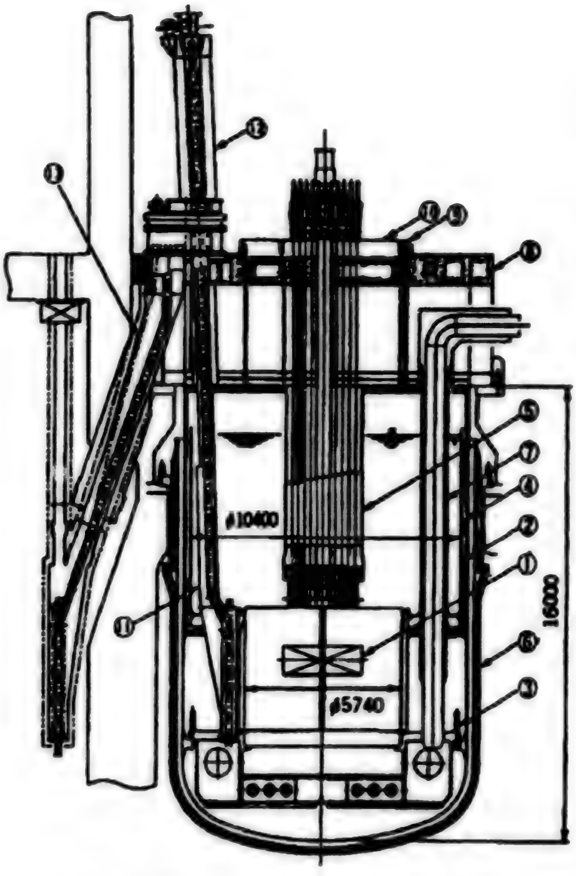
The demonstration reactor, in terms of the technological backing for its design and fabrication, leans heavily on the Monju experience, with meticulous comparisons made with Monju right down to the structural details. In particular, all of the documentation on the Monju plant is in order, including design rationale, basis-constituting R&D results, design books, fabrication, machining, and inspection records, handling instructions, operating manuals, and data from trial operations and full operations. This greatly facilitates the process of drawing from the Monju experience. Furthermore, it makes it possible to better determine the causes of and countermeasures against problems, thus formulating improved measures for the demonstration reactor.

These will each be briefly discussed below.

(1) Mechanical Pumps—In the demonstration reactor, basically that which was used successfully in Monju will probably be adopted. These pumps demand high reliability, so the detailed design work to achieve such reliability, and the know-how and backing, will be based on successful Monju operations.

(2) Intermediate Heat Exchangers—For use in the demonstration reactor, the primary system sodium will be made to flow through the pipes, and the secondary system sodium given a transverse flow, which will be different than the secondary system/counterflow approach used in pipes in Monju. Nevertheless, there are many similarities in the design details, and the Monju design and operation experience will prove determinative in the design of the demonstration reactor.

(3) Steam Generator—For use in the demonstration reactor, the integrated flow-through helical coil configuration is slightly different than the flow-through helical coil-separation configuration used in Monju. Nevertheless, the Monju experience can be drawn from across the board, in terms of design, fabrication, and operation. This will be especially true with regard to the reliability factor, where the results of Monju's trial and full-scale operations will contribute greatly.



Component No.	Name	Material	Quantity
①	Reactor Core		1
②	Reactor Vessel	316FR	1
③	Core Support Structure	SUS304	1
④	Wall Cooling Liner	316FR	1
⑤	Core Top Mechanism	316FR	1 set
⑥	Guard Vessel	SUS304	1
⑦	Inlet Pipeline	SUS304	3
⑧	Loop Deck	Carbon Steel	1
⑨	Large Revolving Plug	Carbon Steel	1
⑩	Small Revolving Plug	Carbon Steel	1
⑪	In-Reactor Chute	SUS304	1
⑫	Fuel Inserter-Retractor	Carbon Steel	1
⑬	Out-of-Reactor Chute	SUS304	1

Figure 6.2 Conceptual View of Demonstration Reactor Structure

(4) Other—Concerning measures to prevent the incursion of air into the primary cover gas system, gas blow-down into equipment, fuel handling system revolving plug, fuel exchangers, sodium equipment and measurement equipment in general, the experiences to date and future successes with Monju will be broadly reflected in the design details, fabrication, and reliability evaluations.

3.3 Structural Integrity

Structural Design: In the demonstration reactor, new materials will be used and the scale will be larger, so the material dynamic behavior and such will be slightly different. Nevertheless, most of the Monju design experience will be reflected. Specifically, this pertains to points subjected to thermal transients and other load conditions, structural evaluation points, dominant evaluation categories, stress-relaxation measures, and the balancing of load, structure, and operation factors.

Earthquake-Resistance Design: In coordinating seismic evaluation models and actual equipment design details, much reliance will be placed on the Monju design, fabrication, and machining experience.

3.4 Operation, Maintenance

Rational methods of operating and maintaining a fast breeder reactor power plant will be developed through the Monju experience. The Monju experience will be reflected in the software and hardware used in the demonstration reactor, in the form of more sophisticated measurement and control equipment and remote-control repair equipment. Concerning ISI systems, advanced environmentally friendly mechanisms, inspection techniques, and remote-control techniques, involving many developmental elements, were adopted for Monju, thereby providing for future development and application. The technology currently achieved and the development experience of the past will be highly useful in developing technology for the demonstration reactor.

3.5 Other

The Monju experience will also be invaluable for the demonstration reactor in terms of getting the proper permits and approvals, and in drafting R&D plans. This involves the nuclear reactor installation permit, construction permits, preliminary inspections, and other authorizations required for a fast breeder reactor.

4. Tool for Demonstrating New Technology

The Monju reactor produces roughly seven times the heat output of the Joyo reactor. This makes Monju not only a vastly larger facility than the large test facility at the Oarai Engineering Center, but also an excellent research facility exhibiting the characteristics of an FBR. Accordingly, there are plans to utilize Monju as a site for developing, applying, improving, and diversifying new technologies that will enhance the safety and economy of future reactors, and as an R&D platform offering irradiation capabilities and other plant features. Furthermore, since it is given a pivotal position as a plutonium-using system, it will be used for project achievement, with a view to international cooperation, and with its activities closely coordinated with the development divisions at the Oarai Engineering Center and Tokai facility.

4.1 Platform for Enhancing FBR Technology

The advances being made in information science and technology are amazing. For example, malfunction diagnostic systems are being developed, operational safety is being enhanced by the application of hierarchical control systems and nonlinear control systems, and conservation support systems are being developed. Not only do these systems provide data and advice to the operator, but are leading to the development of the "intelligent plant" which will to some extent be able to make its own decisions.

R&D successes aimed at improving fuels were also implemented in the Monju reactor core, the fuel cycle cost was reduced by achieving higher burn rates, and operating rates were improved by making the operating cycle longer.

Moreover, there are plans to enlarge reliability databases and perform living PSA-based analyses, reflecting the results in operating procedures, to achieve "worry-free plants" having high reliability, to exploit the characteristics of fast reactors to further reduce radioactivity and discharge into the environment, and to reduce worker exposure. Other plans that are to be implemented include the establishment of the "tough plant" concept, which would naturally expire if trouble developed, the development of remote inspection techniques, high-level ISI technology, and preventative maintenance technology so as to phase in the maintenance-free concept.

4.2 Development of Technologies Which Exploit Plant Features

In the Monju reactor, neutron flux is high, giving it outstanding irradiation capacity potential. In subsequent fuel development, it is hoped that the facility will be used as an "irradiation bed" at the aggregate scale. Also possible will be demonstration tests that are closer to actual full-scale conditions, including performance verification of in-reactor neutron detectors and FFDL pre-location verification. Furthermore, it will be useful to modify the way plants are operated, demonstrating system diversity. In the long term, moreover, there are plans to use Monju, as necessary, for demonstrating such revolutionary technologies as large electromagnetic pumps and ductless fuels which require lower reprocessing costs.

4.3 Diversity Targeting R&D

From the perspective of enhancing the features of fast reactors, there is growing interest in developing revolutionary fast-reactor systems, including reactors operated without regard to breeding, reactors exhibiting greater nuclear-nonproliferation characteristics, and super-uranium element disintegration reactors. We believe that it is important that Monju also be used in such development and demonstration efforts.

5. Recycling Technology Optimization

In order to make practical the plutonium cycle, it is necessary to first develop the fast reactor, and then to optimize the overall cycle of spent fuel reprocessing, fuel processing, and peripheral technology. The focus of future fuel processing technology will be mass production, and technology will be developed to resolve the problems therewith. To do so, plutonium fuel development facilities must be effectively utilized. Fast-reactor fuel reprocessing technology will be developed in close coordination with technological development relating to spent fuel from conventional light water reactors. Spent fuel from fast reactors has a higher plutonium enrichment than does spent fuel from light water reactors, and serious technological development efforts are needed, aiming at high burn rates, in reprocessing technology and fuel processing, assuming industrial scales. To that end, tests are being conducted on equipment and processes based on remote control or automation. In high-level radioactive material research facilities (CPFs), using materials irradiated in fast reactors at burning rates at the 100,000-MWd/t level, reprocessing process (sic) hot tests have been conducted on a laboratory scale. In order to promote the development of reprocessing technology for fast-reactor spent fuel at the next stage, a project to build a recycling equipment test facility (RETF) is being promoted. At the RETF, according to plans, hot tests will be conducted on an engineering scale, using spent fuel aggregates from the FBR Monju, the technology used will be demonstrated, and engineering data will be collected for future facilities. The target year for the start of these hot tests is 2000.

In order to achieve plutonium-using systems, these major development efforts must be coordinated, and process and disposal technology perfected for reducing the amounts of radioactivity released and waste material generated. We believe that, in the future, in the context of a nuclear nonproliferation framework, perfecting optimal cycles for peaceful use of nuclear energy will become an important international focus.

In Conclusion

The fast breeder reactor Monju hieved initial criticality at 10:01 a.m. on 5 April 1994.

This event is considered to have given the FBR a central position in the future of nuclear power generation in Japan. This is a major achievement of Donen, which built and operated the experimental reactor Joyo, and engaged in much testing and research which have yielded many successes. Also, the achievement of criticality in this prototype reactor, which heralds the advent of the era of FBR power generation, can be said to signify the establishment of Japan's own basic FBR technology.

There are plans to continue with reactor physics tests, output tests, and other tests to verify plant performance at Monju. Based on these results, and the experience of operating and maintaining the Monju plant, we now can expect developmental work to move along steadily toward the next phases, namely the demonstration reactor, followed by fully operational reactors.

Meanwhile, Monju will begin to operate as a platform for the international exchange of fast-reactor technology. We are already exchanging information and engineers with France, the United Kingdom, Germany, and the United States. Viewed in terms of the course of international cooperation and the current social environment, we believe that it is Japan's duty to make contributions to the world, using the Monju as the current front runner in fast-reactor development.

In order to firmly implant the FBR in society and make it practicable, it is important that we pursue revolutionary technological development, and that we demonstrate the safety and reliability of FBR plants. From this perspective, we hope to see trial operations proceed steadily at Monju, with the emphasis on "safety first," and to aggressively publicize the gains and results therefrom.

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